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Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants

AN AMERICAN NATIONAL STANDARD

The American Society of Mechanical Engineers

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(All references are distributed within the above sections.)

FOREWORD

The American Society of Mechanical Engineers (ASME) Board on Nuclear Codes and Standards (BNCS) and the American Nuclear Society (ANS) Standards Board mutually agreed in 2004 to form the Nuclear Risk Management Coordinating Committee (NRMCC). NRMCC was chartered to coordinate and harmonize standards activities related to probabilistic risk assessment (PRA) between ASME and ANS. A key activity resulting from NRMCC was the development of PRA standards structured around the Levels of PRA (i.e., Level 1, Level 2, Level 3) to be jointly issued by ASME and ANS. In 2011, ASME and ANS decided to combine their respective PRA standards committees to form the ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM).

In 2006, ASME BNCS established the New Reactor Task Group under the Committee on Nuclear Risk Management (CNRM) to evaluate the need for codes and standards to support the design, construction, licensing, and operation of advanced non-light water reactor (non-LWR) nuclear power plants (NPPs). Following the formation of JCNRM, the New Reactor Task Group is now known as the ASME/ANS JCNRM Advanced Non-LWR PRA Standard Writing Group (Non-LWR WG). The charter of the Non-LWR WG is to develop recommendations to JCNRM on requirements for the performance of PRAs for advanced non-LWRs. The expected applications of such PRAs include input to licensing and design decisions such as selection of licensing-basis events and safety classification of equipment, satisfaction of U.S. Nuclear Regulatory Commission (NRC) PRA requirements for advanced non-LWRs, and support of risk-informed applications for advanced non-LWR NPPs. With the concurrence of JCNRM, the Non-LWR WG decided early on that a new PRA standard was needed to support a broad range of applications for advanced reactor designs.

To support a diverse mixture of reactor concepts, including high-temperature gas-cooled reactors, liquid metal-cooled fast reactors, molten salt reactors, microreactors, and small modular reactors, CNRM decided early on to develop this new PRA standard on a reactor-technology-neutral basis using established technology-neutral risk metrics common to existing light water reactor (LWR) Level 3 PRAs. Such risk metrics include frequency of radiological consequences (e.g., dose, health effects, and property damage impacts). To support a wide range of applications defined by the non-LWR stakeholders, the scope of this Standard is very broad and is comparable to a full-scope Level 3 PRA for an LWR with a full range of plant operating states and hazards. Because some of the advanced non-LWR designs supported by this Standard include modular reactor concepts, this Standard includes requirements that support an integrated risk of multi-reactor facilities including event sequences involving two or more reactors or radionuclide sources concurrently.

In 2013, the JCNRM issued a trial use for the pilot application (TUPA) version of this Standard as ASME/ANS-RA-S-1.4-2013. This trial use version was extensively piloted in the development of a number of advanced non-LWR PRAs that were under development and being built around the world. These advanced non-LWR PRA pilots included one to support the licensing of the HTR-PM pebble bed reactor plant in the Republic of China and a modernization of the GE PRISM PRA, which piloted a major fraction of this Standard's technical requirements in 2018. The experience with pilot applications of this Standard has been extended to support the development of the Traveling Wave and Molten Chloride Fast Reactor at TerraPower, the pebble bed HTGR under development at X-Energy, the Versatile Test Reactor being developed for the U.S. Department of Energy, the Fluoride Cooled High Temperature Reactor at Kairos, the eVinci™ Micro Reactor at Westinghouse, and Advanced HTGRs under development in Japan. The purpose of this version of this Standard is to capture the lessons learned from these pilot applications and to incorporate improvements that have been made in other PRA standards that are applicable to advanced non-LWRs.

In preparing the technical requirements in this Standard, the Non-LWR WG made use of applicable source material from PRA standards that have been developed for LWRs including ASME/ANS RA-Sb-2013, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," as well as Trial Use PRA standards developed by ASME and ANS for Low-Power-and-Shutdown PRA, Level 2 PRA, and Level 3 PRA.

This publication, the 2021 edition of Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants, was approved by the ASME Board on Nuclear Codes and Standards and the ANS Standards Board. ASME/ANS RA-S-1.4-2021 was approved by the American National Standards Institute on January 28, 2021.

CORRESPONDENCE WITH THE ASME/ANS JOINT COMMITTEE ON NUCLEAR RISK MANAGEMENT

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Attending Committee Meetings. The JCNRM regularly holds meetings and/or telephone conferences that are open to the public. Persons wishing to attend any meeting and/or telephone conference should contact the Secretary of JCNRM.

ACKNOWLEDGMENTS

This edition of the advanced non-LWR PRA Standard was the result of the dedicated efforts of the following individuals who are responsible for the changes made to the 2013 trial use version of this Standard: Jordan Hagaman, Peiwen Whysall, Matthew Denman, and Matthew Warner at Kairos Power; Marty Sattison, Individual; Andrew Clark and Jamal Mohmand of Sandia National Laboratory; David Grabaskas of Argonne National Laboratory; Dennis Henneke of GE Hitachi Nuclear Energy; Hanh Phan of the U.S. Nuclear Regulatory Commission; and Karl Fleming, KNF Consulting Services, LLC.

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SECTION 1 – INTRODUCTION
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(The text presented in blue font in this standard comprise hyperlinks to enable efficient access to referenced sections and elements, requirements, notes, references, etc.)

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SECTION 1 INTRODUCTION

1.1 OBJECTIVE

This Standard[1] states the requirements for probabilistic risk assessments (PRAs) used to support risk-informed decisions for advanced non-light water reactor (non-LWR) nuclear power plants (NPPs) and prescribes a method for applying these requirements for specific applications. To support the application of this Standard to PRAs for a diverse set of reactor designs such as modular high-temperature gas-cooled reactors (MHTGRs), liquid metal-cooled fast reactors (LMRs), molten salt reactors (MSRs), micro-reactors and small modular reactors based on non-LWR technology, and other advanced non-LWRs, the requirements in this Standard were developed on a reactor-technology-inclusive basis. This Standard supersedes an earlier trial use version ASME/ANS RA-S-1.4-2013 [1-1][2] and incorporates feedback from pilot studies using that version.

1.2 SCOPE

This Standard states requirements for a PRA for advanced non-LWR NPPs. The requirements in this Standard were developed for a broad range of PRA scopes that may include the following:

(a) Different sources of radioactive material both within and outside the reactor but within the boundaries of the plant whose risks are to be determined within the PRA scope selected by the user;

(b) Different plant operating states, including various levels of power operation and shutdown modes;

(c) Initiating events caused by internal hazards, such as internal events, internal fires, and internal floods, and external hazards such as seismic events, high winds, and external flooding. The only hazards explicitly excluded from the scope are releases resulting from purposeful human-induced security threats (e.g., sabotage, terrorism);

(d) Different event sequence end states, including those with no adverse consequences, plant damage states (PDSs), and release categories that are sufficient to characterize mechanistic source terms, including releases from event sequences involving two or more reactors or radionuclide sources;

(e) Evaluation of different risk metrics including the frequencies of modeled PDSs, event sequence families, release categories, risks of off-site radiological exposures and health effects, and the integrated risk of the multi-reactor plant as defined by the selected PRA scope. The risk metrics supported by this Standard are established metrics used in existing light water reactor (LWR) Level 3 PRAs such as frequency of radiological consequences (e.g., dose, health effects) that are independent of reactor technology. Surrogate risk metrics used in LWR PRAs such as core damage frequency (CDF) and large early release frequency (LERF) are not applicable to many non-LWR designs and not used in this Standard;

(f) Quantification of the event sequence frequencies, mechanistic source terms, off-site radiological consequences, risk metrics, and associated uncertainties, and using this information to support risk-informed decisions in a manner consistent with the scope and applications PRA.

Technical requirements are provided in this Standard for different combinations of sources of radioactive material, plant operating states, and hazard groups. In this Standard, the technical requirements for the internal fire hazard group are limited to "at-power" plant operating states as there was insufficient experience in performing internal fire PRAs during low power and shutdown (LPSD) plant operating states to justify the inclusion of such requirements. This exclusion is consistent with supporting LWR PRA standards. Other combinations of hazard groups and plant operating states are, however, supported in this Standard.

It is recognized that for some PRA applications, a PRA that addresses the full set of requirements covered in this Standard may not be required. In addition, for PRAs performed in various stages of design and licensing, especially those PRAs performed prior to the selection of a specific site, the level of detail and completeness of the PRA with respect to the PRA scope, coverage of events and event sequences, plant operating states, hazards, risk metrics, and operational characteristics may be limited in relation to that for typical PRAs for an operating plant. Hence, the scope of the PRA is established by the user in accordance with the intended PRA applications and the availability of information to support each PRA element. In addition, the requirements in this Standard for the level of detail, completeness, and model to plant or design fidelity vary according to the scope and level of detail of design and operational information that is available to support, and is referenced by, the PRA with additional requirements to address assumptions in lieu of as-operated and as-built details.

[1] The current Standard, ASME/ANS RA-S-1.4-2021, is herein referred to as "this Standard."

[2] Numbers in brackets refer to corresponding reference numbers at the end of each section in "Reference(s)."

Hazards explicitly excluded from the scope of this Standard are radionuclide releases resulting from purposeful human-induced security threats (e.g., sabotage, terrorism) and risks associated with accidental radiological exposures to on-site personnel. This Standard applies to PRAs used to support applications of risk-informed decision-making (RIDM) related to design, licensing, procurement, construction, operation, and maintenance of advanced non-LWR NPPs.

This Standard includes requirements for addressing the off-site radiological consequences of event sequences involving the release of radioactive material from the plant. The consequence metrics are common to those found in LWR Level 3 PRAs. The scope of these requirements excludes the consideration of radiological or other consequences to on-site personnel.

1.2.1 Treatment of Hazard Groups

This PRA Standard provides specific requirements for the following hazard groups:

(a) internal events;
(b) internal floods;
(c) internal fires;
(d) seismic events;
(e) high winds;
(f) external floods;
(g) other internal and external hazards.

In addition to providing technical requirements for PRA modeling of various hazards, this Standard provides requirements for screening and conservative analyses of internal and external hazards.

Many of the technical requirements for the internal events hazard group are fundamental requirements for performing a PRA for any hazard group and are therefore relevant to the remaining hazard groups in this Standard. The relevant internal events technical requirements are incorporated in the requirements of other hazard groups by reference to the supporting internal events requirements. Such back references represent that a fundamental understanding of the plant response to a reasonably complete set of initiating events as defined in the internal events PRA model provides the foundation for modeling the impact of various hazards on the plant. Hence, requirements defined for the internal events hazard group apply to all the hazard groups within the scope of the PRA. When the internal events PRA models are adapted for use in the treatment of other hazard groups, the potential for new initiating events and event sequences that may impact multiple reactors and sources of radioactive material needs to be considered.

1.2.2 Hazards and Initiating Events

In using this Standard, it is necessary to understand the relationships among hazard, hazard group, hazard event, and initiating event, which are defined in Section 2. "Hazard" is the specific phenomenon that puts the plant at risk. "Hazard group" refers to collections of similar hazards that are assessed in the PRA using a common approach, common methods, and common data. So, a hazard group may consist of only a single hazard (e.g., internal fires or seismic events) or multiple hazards [e.g., internal events group, which includes transients and reactor coolant system boundary (RCB) breaches; high winds, which include all hazards associated with either hurricanes, tornadoes, or straight winds]. In this context, hazard is the phenomenon; hazard event is an occurrence of the phenomenon that can result in a plant trip and possibly other damage when the plant is at-power or result in the loss of a key safety function during nonpower operations, and the initiating event is the specific plant perturbation that challenges plant control and safety systems.

In general, there is a range of hazard events associated with any given hazard, and for analysis purposes, the range can be divided into bins characterized by their severity. Hazard events of different severity can result in different initiating events. Examples of the overall concepts described above are as follows:

(a) A hazard is an earthquake.

(b) A hazard group is a tornado, which may generate (1) high rotational winds, (2) atmospheric pressure drop, and (3) impact of missiles.

(c) A hazard event is an earthquake of a given intensity, defined by the associated peak ground acceleration (e.g., 0.1 g, 0.2 g, 0.3 g, 0.4 g, 0.5 g with g = gravitational acceleration), spectral shapes, and time histories.

(d) An initiating event is a manual plant trip, which is typically generated by a 0.1 g earthquake. Also, an initiating event is a loss of off-site power, which is typically generated by higher than 0.3 g earthquakes.

As another example, for internal events,

(a) RCB breaches may be identified as a generic type of hazard.

(b) The specific hazard events would be RCB breaches of different sizes and locations, etc.

(c) Small RCB breach leading to plant trip on low reactor coolant system (RCS) pressure would be the initiating event for the small RCB breach hazard event.

Note that initiating events are defined in terms of the plant disturbance, and a given initiating event may be induced by, for example, an internal event, internal fire or flood, or external hazard.

1.3 REQUIREMENTS FOR DIFFERENT DESIGN-LIFE CYCLE STAGES

This Standard applies to PRAs used to support applications of RIDM related to design, licensing, procurement, construction, operation, and maintenance of advanced non-LWR NPPs. These requirements are written for PRAs that are performed, maintained, and upgraded at different times during the plant design and operational life cycle, including pre-licensing design support, design certification (approval by a regulatory body), construction permit application, and combined operating license (a licensing process used by the U.S. Nuclear Regulatory Commission) application, as well as for PRAs performed throughout plant operation until decommissioning.

If not caveated in the requirement, the scope of a given requirement is applicable equally to all stages of the plant life cycle. However, not all requirements are applicable to all stages of the plant life cycle. Furthermore, the PRA analyst may intentionally limit the scope of the PRA given the state of the design. The basic approach to applying requirements to PRAs that may be performed in different stages of the plant design, licensing, and operation is based on the principle that the scope, level of detail, and point of reference for judging the fidelity of the PRA models will vary according to the plant life cycle stage. Hence, the scope and level of detail and completeness of the PRA models are expected to progressively increase, and the point of reference for the supporting design and operating details is expected to change as PRAs are performed and upgraded at various stages of the plant design and operating life cycle. At early stages of the design, it is expected that the scope of the PRA will include little or no treatment of hazards beyond internal events and plant operating states beyond that needed for "at-power PRA" due to a lack of design and operational data. The scope and level of detail of the PRA models cannot exceed that of the supporting design and operational information available when the PRA is performed. In addition, some PRA requirements that are appropriate for an operating plant or a plant already constructed may not be achievable or appropriate for a PRA on a plant in various design and licensing stages. Where possible, the requirements in this standard indicate the appropriate plant life cycle stage that the requirement applies toward. These suggestions may not be universally applicable to every licensee's development pathway but should serve as a useful guide to the analyst and the peer review teams. All of these considerations were addressed in the formulation of the requirements presented in this Standard. This was done by specifying that the PRA scope is selected by the user, and the inclusion of specific requirements to document the limitations of the PRA models due to a lack of design or operational details. As a result, assumptions made in lieu of such details can be reviewed in subsequent PRA upgrades and to ensure that such limitations are taken into account in the supported PRA applications.

1.4 STRUCTURE FOR PRA REQUIREMENTS

1.4.1 PRA Elements

The requirements of this Standard are organized by 18 PRA elements that comprise a full-scope PRA. The PRA elements define the scope of the analysis for each section of this Standard. This Standard specifies technical requirements for the PRA elements listed[3]:

(a) Plant Operating State Analysis (POS);
(b) Initiating Event Analysis (IE);
(c) Event Sequence Analysis (ES);
(d) Success Criteria Development (SC);
(e) Systems Analysis (SY);
(f) Human Reliability Analysis (HR);
(g) Data Analysis (DA);
(h) Internal Flood PRA (FL);
(i) Internal Fire PRA (F);
(j) Seismic PRA (S);
(k) Hazards Screening Analysis (HS);
(l) High Winds PRA (W);
(m) External Flooding PRA (XF);
(n) Other Hazards PRA (O);
(o) Event Sequence Quantification (ESQ);
(p) Mechanistic Source Term Analysis (MS);
(q) Radiological Consequence Analysis (RC);
(r) Risk Integration (RI).

The relationship among these elements and the PRA elements defined in supporting PRA Standards is explained in Section 4. Some of the PRA elements are divided into subelements to help organize the formulation of technical requirements.

The relationship between the PRA elements and the scope of the PRA with respect to hazard groups is shown in Table 1.4-1. In particular, it is important to note that the requirements specified in this Standard for the PRA elements Plant Operating State Analysis, Initiating Event Analysis, Event Sequence Analysis, Success Criteria Development, Systems Analysis, Human Reliability Analysis, Data Analysis, and Event Sequence Quantification apply to both internal and external hazards and associated hazard groups and all sources of radioactive material in the scope of the PRA. Depending on the scope of the PRA, these elements may also address the integrated risk of a multi-reactor plant. Depending on the specific scope of the PRA, the PRA elements Mechanistic Source Term Analysis, Radiological Consequence Analysis, and Risk Integration may be applied to internal events only from a single plant operating state and source of radioactive material, or to any combination of radionuclide sources, plant operating states, and hazard groups, as well as to the integrated risk of a multi-reactor plant.

[3] The letter codes listed after the names of the technical elements are not acronyms but rather letter codes used to identify specific technical (high level and supporting) requirements.

Table 1.4-1 PRA Elements and Hazard Groups Addressed in This Standard

PRA Elements	Scope of Hazard Groups		
	Internal Events	Other Internal Hazards	External Hazards
Plant Operating State Analysis (POS)	X	X	X
Initiating Event Analysis (IE)	X	X	X
Event Sequence Analysis (ES)	X	X	X
Success Criteria Development (SC)	X	X	X
Systems Analysis (SY)	X	X	X
Human Reliability Analysis (HR)	X	X	X
Data Analysis (DA)	X	X	X
Internal Flood PRA (FL)		X	
Internal Fire PRA (F)		X	
Seismic PRA (S)			X
Hazards Screening Analysis (HS)			X
High Winds PRA (W)			X
External Flooding PRA (XF)			X
Other Hazards PRA (O)			X
Event Sequence Quantification (ESQ)	X	X	X
Mechanistic Source Term Analysis (MS)	X	X	X
Radiological Consequence Analysis (RC)	X	X	X
Risk Integration (RI)	X	X	X

1.4.2 High Level Requirements

A set of objectives and High Level Requirements (HLRs) is provided for each PRA element in Section 4. The HLRs set forth the minimum requirements for a technically acceptable PRA, independent of an application. The HLRs are defined in general terms and present the top-level logic to derive more detailed Supporting Requirements (SRs). The general terms used for HLRs represent not only the diversity of approaches that have been used to develop the existing PRAs but also the need to accommodate future technological innovations, as well as to support PRAs being performed in different stages of the plant design and licensing life cycle.

1.4.3 Supporting Requirements

A set of SRs is provided for each HLR (which is provided for each PRA element) in Section 4.3.

This Standard is intended for a wide range of applications that require a corresponding range of PRA capabilities. PRA applications vary with respect to the following factors:

(a) type of reactor;

(b) which stage in the plant design and life cycle the PRA is developed to support, which risk metrics and risk significance criteria are employed;

(c) which decision criteria are used;

(d) the extent of reliance on the PRA results in supporting a decision; and

(e) the degree of resolution required for the factors that determine the risk significance of the subject of the decision.

In developing the different portions of the PRA model, not every item (e.g., system model) will require the same level of detail, same degree of plant-, site- or design-specificity, or the same degree of realism.

For each SR, the minimum requirements necessary to meet Capability Category I (CC-I) and Capability Category II (CC-II) are defined. Some SRs apply only to one Capability Category, and some extend across both Capability Categories. When an SR spans both Capability Categories, it applies equally to each Capability Category. When necessary, the differentiation between Capability Category is made in other associated SRs.

Although the range of capabilities required for each portion of the PRA to support a PRA application falls on a continuum, two levels are defined and labeled CC-I and CC-II so that requirements can be developed and presented in a manageable way. Table 1.4-2 describes, for three principal attributes of PRA, the bases for defining the Capability Category. Table 1.4-2 was used to develop the SRs for each HLR.

Table 1.4-2 Bases for PRA Capability Categories

Attributes of PRA	Capability Category I	Capability Category II
1. Scope and level of detail: The degree to which the scope and level of detail of the plant design, operation, maintenance, and physical phenomena relevant to the plant design are modeled.	Resolution and specificity are sufficient to identify the relative importance of the contributors at the hazard group, initiating event group, and functional or systemic event sequence level including associated human failure events, relevant physical phenomena, and release characteristics [Notes (1), (2)].	Resolution and specificity sufficient to identify the relative importance of the risk-significant contributors at the hazard group, initiating event group, functional and systemic event sequence, and basic event level including associated human failure events, and relevant physical phenomena and release characteristics [Notes (1), (2)].
2. Plant-specificity: The degree to which plant-, site- or design-specific information is incorporated such that the as-built and as-operated plant, or as-designed plant, is addressed.	Use of generic data/models is acceptable except for the need to account for the unique design and operational features on the assessment of risk.	Plant-, site- or design-specific data/models are used for the risk-significant contributors to the extent feasible.
3. Realism: The degree to which realism is incorporated such that the expected response of the plant is addressed.	Departures from realism will have moderate impact on the conclusions and risk insights as supported by the state-of-practice. A departure from realism should be in the conservative direction. [Note (3)].	Departures from realism will have small impact on the conclusions and risk insights supported by the state-of-practice [Note (3)].

MANDATORY NOTES:

(1) The plant-, site- and/or design-specificity and realism attributes will be used to determine that the hazard scenarios are evaluated in a manner consistent with the other contributors, subject to the limitations imposed by the differences in treatments of each hazard.

(2) The definitions for CC-I and CC-II are not meant to imply that the scope and level of detail include identification of all components and human actions, but only those needed for the function of the system being modeled to the extent that function is important to assessing plant risk as defined in the context of this Standard.

(3) Differentiation from moderate to small is determined by the extent to which the impact on the conclusions and risk insights could affect a decision under consideration. This differentiation recognizes that the PRA would generally not be the sole input to a decision. A moderate impact implies that the impact (of the departure from realism) is of sufficient size that it is likely that a decision could be affected; a small impact implies that it is unlikely that a decision could be affected.

The delineation of the Capability Categories within the SRs is generally that the degree of scope and level of detail; the degree of plant-, site-, or design-specificity; and the degree of realism (i.e., the depth of the analysis) increase from CC-I to CC-II. As the Capability Category increases, the depth of the analysis required also increases. In other cases, increasing the depth of analysis may result in a decrease in the risk, such as when a conservative assumption is refined to be more realistic (e.g., changing from conservative success criteria to more realistic success criteria).

The boundary between these Capability Categories can be defined only in a general sense. When a comparison is made between the capabilities of any given PRA and the SRs of this Standard, it is expected that the capabilities of a PRA's elements or portions of the PRA within each of the elements will not necessarily all fall within the same Capability Category, but rather will be distributed between both Capability Categories.

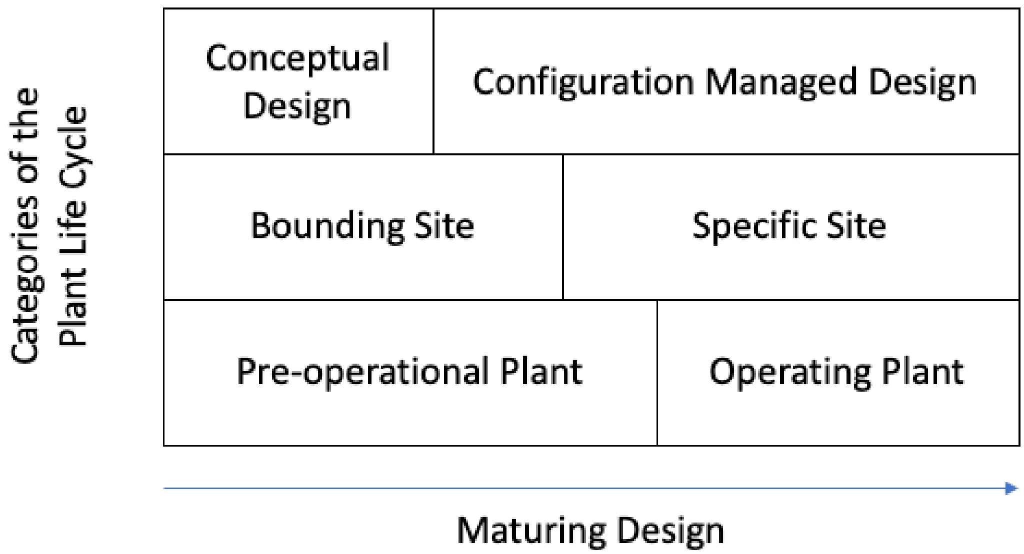
There may be PRA technical elements or portions of the PRA within the elements that fail to meet the SRs for either of these Capability Categories. CC-I requirements should result in a model that is capable of identifying the most risk-significant event sequences at a functional or systemic level. CC-II will provide a realistic assessment of risk. Further, the SRs have been written so that, within a Capability Category, the interfaces between portions of the PRA are consistent (e.g., requirements for event trees are consistent with the definition of initiating event groups).

When a specific PRA application is undertaken, judgment is needed to determine which Capability Category is needed for each portion of the PRA, and hence, which SRs apply to the PRA applications.

The Technical Requirements Section of each respective technical element of this Standard also specifies the required documentation to ensure traceability of the analysis.

Not all SRs apply to all plant life cycle stages. Where possible, this Standard states when SRs apply to the PRA. Figure 1.4-1 shows four stages of the development and maturation of a design that this Standard uses to suggest applicability of an SR. For example, an analysis of a design across a range of sites cannot be as site-specific as a design analyzed on a specific site. As such, certain SRs that require a high degree of site specificity may not apply to a PRA before site selection. Once the site is selected, those SRs will apply and thus need to be satisfied.

Fig. 1.4-1 Major Phases of a New Reactor Development for Which Applicability of Various SRs May Change in This Standard.



Some SRs can be achievable only with abundant plant- or site-specific information. These SRs are broken into two requirements, one requirement providing a success pathway for a design with limitations in details that are to be expected due to an early stage in the plant life cycle and a second requirement that will expect the incorporation of all plant- or site-specific data. When developing these recommendations, this Standard assumes the following:

(a) Conceptual Design versus Configuration Managed Design – PRAs performed in the conceptual design phase likely do not benefit from a configuration management system. The Configuration Control technical element is likely not achievable at this phase.

(b) Bounding Site (e.g., bounding a range of sites) versus Specific Site – A bounding site analysis is conducted before a specific site is selected. Assumptions associated with grading changes to the site and soil interactions with to-be-constructed structures may make some SRs unachievable at the bounding site stage that would be achievable after a site is selected and construction begins.

(c) Pre-operational Plant versus Operating Plant – Some information (e.g., operational experience, procedures, evaluation plants) is not expected to be available until shortly before or after the plant begins operation.

If the proper information exists for a design to meet an SR, before that information is expected to exist when the SRs were written, the PRA analyst has the flexibility to scope the more aggressive plant- or site-specific requirements into the early stage PRA (e.g., if operating procedures are available before plant operation, the PRA can use the operating plant requirement for incorporating procedures instead of the pre-operational requirement that bypasses the incorporation of procedures).

The SRs specify what to do rather than how to do it, and in that sense, specific methods for satisfying the requirements are not prescribed. Nevertheless, certain established methods were contemplated during the development of these requirements. Alternative state-of-practice methods and approaches or newly developed methods for meeting the requirements of this Standard may be used if they provide results that are equivalent or superior to the methods usually used and they meet the HLRs and the SRs presented in this Standard. Requirements for newly developed methods are provided in Section 7. The use of any particular method for meeting an SR shall be documented and shall be subject to review by the peer review process described in Section 1.7. All Notes and Commentaries, which follow many SRs, are nonmandatory. In addition, any example in the SR body or any nonmandatory appendix or note is not to be considered to be the only way to address a SR.

1.5 APPLICABILITY OF PRA TECHNICAL ELEMENTS

The use of a PRA and the selection of the technical requirements and Capability Categories that need to be met will differ for different PRA applications. A part of a PRA model might be the event sequences involving a specific set of initiating events or system responses (e.g., event sequences involving station blackout), and hence, PRA modeling items should not be confused with PRA technical elements, which are a means of organizing the technical requirements of this standard. Section 3 describes the activities to determine whether a PRA has the capability to support a specific application of RIDM. Two different PRA Capability Categories are described in Section 1.4. PRA capabilities are evaluated for each associated SR, rather than by specifying a Capability Category for specific modeling items or the whole PRA.

1.6 PRA CONFIGURATION CONTROL

Section 5 states the requirements for configuration control of a PRA (i.e., maintaining and upgrading a plant-specific PRA) such that the PRA represents the as-designed or as-built, as-operated facility to a degree sufficient to support the PRA application for which it is used.

1.7 PEER REVIEW REQUIREMENTS

Section 6 states the general requirements for a peer review to determine whether the PRA methodology and its implementation meet the technical requirements in Section 4. Peer review requirements that are specific to each PRA element are presented with the associated technical requirements in Section 4.

1.8 ADDRESSING DIFFERENT PRA SCOPES

The scope and level of detail of a PRA that is developed during the pre-operational stage or for a bounding site must account for the scope and level of detail of the design, operation, maintenance, and siting information that is available for the plant. PRAs developed during the conceptual design may be limited to a subset of the possible plant operating states and may include the internal event hazard group. The PRA application process described in Section 3 calls for the user of this Standard to select those technical requirements and risk significance criteria that are applicable to the scope of the PRA and the PRA applications applicable to that stage of design.

The technical requirements to determine the technical adequacy of a PRA for different sources of radioactive material, plant operating states, hazard groups, and risk metrics to support applications are presented in Section 4. The approaches to modeling the plant damage resulting from these different PRA scopes vary in terms of the degree of realism and the level of detail achievable by the state-of-the-art. For example, there are uncertainties that are unique to the modeling of the different sources and hazards within different plant operating states and their effect on the plant, and the assumptions made in dealing with these uncertainties can lead to varying degrees of conservatism in the estimates of risk. Furthermore, because the analyses can be resource intensive, it is normal to use screening approaches to limit the number of detailed scenarios to be evaluated and the number of mitigating systems credited while still achieving an acceptable evaluation of risk. These screening approaches may differ among different sources of radioactive material, plant operating state, and hazard group, but the basic approach to screening is consistent with that presented in Section 1.10.

For many applications, it is necessary to consider the combined impact on risk from those radionuclide sources, plant operating states, and hazard groups for which it cannot be demonstrated that the impact on the decision being made is insignificant. This can be done by using a single integrated risk model that combines the PRA models for the different sources, states, and hazard groups or by combining the results from separate models. In either case, when combining the results from the different sources, states, and hazard groups, it is essential to account for the differences in levels of conservatism and levels of detail so that the conclusions drawn from the results are not overly biased or distorted. To support this objective, this Standard is structured so that requirements for the analysis of the PRA results, including identification of risk-significant contributors, identification and characterization of sources of uncertainty, and identification of assumptions are included in each element of the PRA scope separately.

In some cases, the requirements for developing a PRA model in a specific hazard group (e.g., internal floods) refer back to the requirements that are common to all internal and external hazards (e.g., Systems Analysis). The requirements for PRA elements common to all hazards should be applied to the extent needed given the context of the modeling of each hazard group. Many of the requirements that differentiate between Capability Categories, either directly or by incorporating the requirements for PRA elements common to all hazards, do so on the basis of the treatment of risk-significant contributors and risk-significant event sequences/cutsets for the source of radioactive material, plant operating state, or hazard group being addressed. Because there are differences in the way the PRA models for each specific source, plant operating state or hazard group are developed, the requirements are best treated as being self-contained for each source, plant operating state and hazard group separately when determining risk-significant contributors and risk-significant event sequences/cutsets. In other words, these are identified with respect to the release category frequencies (RCFs) modeled in the PRA for each source, plant operating state, and hazard group separately as well as for the total integrated risk across all the sources, plant operating states, and hazard groups, as may be required for the application. While there is a need in some applications to assess the significance with respect to the total integrated risk from all modeled sources, plant operating states, and hazard groups, such an integrated assessment has to be done with a full understanding of the differences in conservatism and level of detail introduced by the modeling approaches for the different sources, plant operating states, and hazard groups, as well as within each of these key elements of the PRA scope.

To determine the Capability Category at which the SRs have been met, it is necessary to have a definition of the term "risk-significant" whose requirements are discussed for the Risk Integration technical element in Section 4. Consequently, the term "risk-significant" is used in various definitions in this Standard and is thereby explicitly incorporated into specific SRs. Generally, the philosophy used in CC-II ensures a higher level of realism for risk-significant contributors. This manifests itself in SRs related to the scope of plant-specific data, detailed human reliability analysis (versus screening values), common cause failure treatment, documentation, and others.

The only consequence of not meeting this Standard's definition of "risk-significant" for a specific SR is that the PRA would not meet CC-II for that SR. Thus, in the context of an application, if a source, plant operating state, or hazard group is a small contributor, it should be acceptable to meet CC-I by using conservative human error probabilities, not using plant-specific data for equipment reliability, etc. The applicable portion of the PRA will simply be considered as meeting CC-I for that specific SR for that hazard group.

Additionally, from a practical standpoint, PRA models are generally developed for a specific combination of radionuclide source, plant operating state, and hazard group basis (e.g., a PRA model for event sequences initiated at full power with releases from a reactor core). While the models for different sources, plant operating states, and hazards may be integrated into a single model, the initial development for each combination may be done separately. In CC-II, this Standard strives to ensure that the more risk-significant contributors to each source, plant operating state, and hazard group are understood and treated with an equivalent level of resolution, plant-specificity, and realism, so as not to skew the results for that hazard group. The definitions also acknowledge that there may be cases where the proposed quantitative assessment process is inappropriate (e.g., the hazard group risk is very low, or bounding methods are used). Hence, depending on the context, significance is defined both in terms of specific sources, plant operating states, and hazard groups and with respect to the total integrated risk. This Standard includes requirements for selecting the criteria to be used to establish risk significance as discussed in Section 1.9.

To summarize, the definitions that use the term "risk-significant" simply help to define how much realism is necessary to meet CC-II of some SRs. By contrast, the concept of risk significance is used for risk-informed decisions supported by the PRA. In the context of this Standard, the decisions on applying these definitions and/or defining what is important to a decision in an absolute sense would be addressed in the Risk Assessment Application Process (see Section 3).

1.9 DETERMINING RISK-SIGNIFICANT ITEMS

A critical aspect in developing a state-of-practice PRA model (e.g., CC-II requirements) is to have a realistic PRA that is developed in sufficient scope and level of detail to represent the as-built and as-operated or as-designed plant. Generally, preliminary PRA modeling will use simpler models and conservative assumptions when needed (e.g., generic data, point estimates, screening values). This simplification may impact multiple portions of the PRA modeling (e.g., risk contributors, basic events, SSCs, event sequences). Therefore, initial quantification may not produce a realistic risk profile representative of the as-built and as-operated plant. However, applying CC-II requirements to every aspect of the PRA model may not increase the realism of the model; therefore, it may not be necessary to apply every requirement to the same Capability Category for every aspect of the model.

Risk significance is often used in the iterating refining process used to build a PRA model. Hence, the initial PRA model, possibly built with simplifying assumptions that may impact multiple portions of the PRA model, is generally reviewed and modified as needed to become more realistic. This type of iteration is performed until the PRA model represents a realistic risk profile of the plant, to the extent practical according to the state-of-practice. Consequently, the focus should be on increasing the realism of those aspects of the PRA that have the potential to significantly impact the model results.

Generally, this iterative process involves the major design stages, as discussed below. By the end of this iteration, a sufficiently realistic PRA model with the appropriate risk-significant insights is expected to be consistent with the state-of-practice.

First, a preliminary PRA model is constructed. This may include modeling of contributors (e.g., radionuclide sources, hazard groups, initiating events, event sequences, equipment failures, human failure events, etc.) with varying levels of details, including simplifying assumptions. In developing these preliminary PRA models, the outputs are obtained and preliminary risk contributors are identified.

Second, specific technical requirements that may apply to a specific risk contributor are identified to understand what additional refinement actions may be needed to generate a more realistic set of risk-significant contributors.

Third, the refined model is re-quantified to assess whether new risk-significant modeling inputs are identified. The number of iterations needed is highly dependent on the results and insights produced by the PRA model and the refinements needed. It is possible that each individual iteration may highlight items that are less or more risk-significant, and the goal is to achieve a realistic representation of the plant to the extent possible as supported by the state-of-practice.

This Standard includes the option to use either relative or absolute criteria for establishing risk significance depending on the PRA application. Risk significance may be determined for any well-defined item or items in a PRA model including basic events; initiating events; event sequences; event sequence families; release categories;

structures, systems, and components (SSCs); human failure events; functions, hazards; and sources of radioactive material within the scope of the PRA model. As noted in Section 3, PRA Application Process, it is the responsibility of the user to define the scope and level of detail of the PRA and the applicable technical requirements and Capability Categories for these requirements for a given stage of design and for the intended application.

Users of this Standard may select relative or absolute criteria to establish risk significance, depending on the intended application of the PRA.

(a) Relative risk significance criteria are selected to support PRA applications where baseline risk is calculated.

(b) Absolute risk significance criteria are selected to support PRA applications where the acceptability of the risk is evaluated against fixed risk targets.

Examples of applications using absolute criteria are those where the results and insights from the PRA are used as input to the evaluation of design options; selection of licensing basis events; safety classification of SSCs); definition of performance requirements for SSCs; and evaluation of defense-in-depth adequacy.

As explained in Section 1.10, screening criteria for managing the scope and complexity of the PRA models are defined in such a manner to ensure that risk-significant items are not screened out using either relative or absolute risk significance criteria. For any given application of the PRA, it is the responsibility of the user to determine and justify the appropriate roles of absolute and relative risk significance criteria.

1.9.1 Risk Metrics for Evaluation of Risk Significance

The risk metrics used in this Standard for evaluation of risk and risk significance are reactor technology inclusive and explicitly include the quantification of both the frequency and consequences of event sequences modeled in the PRA. The risk metrics are considered technology inclusive because they use consequence metrics such as radiological doses and health effects to the public that are independent of the reactor technology. These metrics include and are derived from event sequence frequencies, radiological release source terms, and off-site radiological consequences. Event sequences are grouped into event sequence families having similar initiating event, plant response, and mechanistic source term. Event sequences may be grouped into release categories based on similarity of the release to facilitate efficiencies in performing the consequence analysis.

The LWR risk metric CDF is not used because it is not applicable to non-LWRs as defined in LWR PRA standards. Large release frequency and LERF are not used because they are not necessary in this Standard given the requirement to quantify source terms and radiological consequences. User-defined intermediate metrics may be used if justified by the user. The selection of risk metrics for this Standard facilitates the modeling of event sequences that may involve two or more reactors or non-reactor radiological sources and the integration of a total risk profile that addresses all plant operating states and hazard groups [1-1].

The risk metrics used to establish risk significance of event sequences include the product of frequency and consequence, exceedance frequency versus consequence, individual risks for the population surrounding the plant, and plots of frequencies and consequences of event sequence families against a frequency consequence target. Consequence metrics include site boundary dose, population man-rem exposure, and early and latent health effects from which the user may select to support a given application.

This Standard recognizes PRA applications in which there is a need for aggregating the risk over the individual sequences modeled in the PRA: risk may be aggregated to assess the total integrated risk of the plant as captured within the scope of the PRA or may be aggregated separately for specific combinations of radionuclide source, plant operating state, and hazard, or other well-defined portions of a PRA model. When risk is aggregated across different sources, plant operating states, and hazards, there are requirements in this Standard to address technical issues associated with risk aggregation.

1.9.2 Relative Risk Significance Criteria

A practical implementation on how to identify actions for further refinement of a PRA model can be based on relative quantitative criteria (i.e., the contributor under consideration that contributes above a percentage to the overall risk is included in the final quantified risk). For measuring relative risk significance against the total integrated risk, a threshold of 95% is reasonable to assume such that, if the remaining 5% contribution is not refined, it will not affect the integrity of the model and, therefore, will not impact the results and insights. In addition, an individual contributor could be sufficiently important by itself to the risk insights profile. A practical approach to address this condition is an assessment of the relative importance of the individual contributor via thresholds expressed as a percent contribution to risk or by use of thresholds for risk importance measures such as Fussell-Vesely (FV) or risk achievement worth (RAW). These thresholds should be set such that the calculation of the integrated baseline risk metric can be supported. SRs for the selection of relative risk criteria for such applications are included in SR RI-A1.

Table 1.9-1 specifies the quantitative criteria that may be used in determining relative risk significance for the various modeling items (contributors). Justification for the use of alternative criteria is required by the requirements in the Risk Integration element. Such alternative criteria are required to be peer reviewed.

Once potential risk-significant contributors are defined, the refined model is re-quantified to assess whether new modeling relative risk-significant modeling inputs are identified. The number of iterations needed is highly dependent on the results and insights produced by the PRA model and the refinements needed. It is possible that each individual iteration may highlight items that have a relative risk significance that is less or more relative risk-significant than the previous iteration. The goal is to achieve a realistic overall risk profile of the plant where the relatively risk-significant contributors and insights are identified to the extent possible as supported by the state-of-practice.

The numerical thresholds used to identify risk-significant items should be considered in the context of the resolution of the quantification process. In some cases, especially for external hazards such as seismic where the quantification process may be impacted by widespread impacts and the failure of the rare event approximation, a F/V importance of 0.005 is not meaningful or potentially misleading because of the bias toward a specific hazard interval (which may have higher truncation and less cutsets, and may generate a distorted importance to specific basic events). In such cases, a higher numerical threshold may be used and justified (e.g., F/V importance of 0.02) without distorting the insights from the analysis.

Table 1.9-1 describes how risk significance is determined for the different types of modeling items (contributors). In applying the criteria in Table 1.9-1, it is important to understand that the SRs described in the various elements of this Standard are written under the assumption that the specific hazard addressed in an element is the most important hazard in the PRA. In other words, the SRs written for Internal Fire PRA are written assuming that fire is the most important hazard in the plant. At the same time, the SRs written for Seismic PRA are written assuming that the seismic hazard is the most important hazard at the site. Depending on the PRA application, this is not always the case. For risk-informed applications that are based on the total plant risk, the concept of risk significance for a specific item in a specified hazard may be calibrated (weighted) by the importance of that hazard in the overall risk profile. While any specific guidance on this is left to the specific application implementation guidance, a relative definition of risk significance can be specifically defined for the specific application. In this context, if, for example, the seismic hazard only contributes less than 1% to the overall plant risk, a Seismic PRA developed in support to an application that uses total risk profile may be able to justify a higher risk-importance threshold, for example, for fragilities that are expected to be refined and realistic. Such refinements are required to be documented and subject to peer review.

This said, while a hazard group may not be risk-significant to the overall plant risk profile, an understanding of what is risk-significant to that hazard group may be beneficial to gain the insights for that hazard group for specific PRA applications.

Table 1.9-1 Relative Risk Significance Criteria

Item	Relative Risk Significance Determination
Risk-significant basic event, relative	A basic event that contributes significantly to baseline risk. It is defined as any basic event that has an FV importance greater than 0.005 or a RAW importance greater than 2 where the importance is normalized against the baseline total integrated risk or risk of a specific combination of source of radioactive material, hazard, and plant operating state.
Risk-significant function, relative	An event or element of a PRA model that represents the performance of a safety function that contributes significantly to risk. For this version of the Standard, the aggregate percentage for the set is 95%, and the individual event sequence or event sequence family percentage is 1% of the total integrated risk or risk of a specific combination of source of radioactive material, hazard, and plant operating state.
Risk-significant contributor, relative	Implies a significant contributor to a given baseline risk metric that is expressed as the total integrated risk or risk of a specific combination of source of radioactive material, hazard, and plant operating state. Contributors may be defined in terms of a basic event, piece of equipment, an a human failure event (HFE), scenario, initiating event, event sequence, event sequence family, release categories, or other defined modeling item of the PRA model.
Risk-significant event sequence or event sequence family, relative	An event sequence or event sequence family that, when rank-ordered by decreasing frequency, contributes a specified percentage of the baseline risk, or that individually contributes more than a specified percentage of the risk. For this version of the Standard, the aggregate percentage for the set is 95%, and the individual event sequence or event sequence family percentage is 1% of the total integrated risk or risk of a specific combination of source of radioactive material, hazard, and plant operating state.
Risk-significant plant operating state, relative	A plant operating state that is associated with one or more risk-significant event sequences families. The risk significance of a plant operating state is the total risk of the associated risk-significant event sequences and may be expressed in terms of the total integrated risk, or the risk of a specific combination of source of radioactive material, hazard, and plant operating state.
Risk-significant cutset, relative	A cutset resulting from the analysis that, when rank-ordered by decreasing frequency, sums to a specified percentage of the total integrated baseline risk or risk of a specific combination of source of radioactive material, hazard, and plant operating state, or that individually contributes more than a specified percentage of risk. For this version of the Standard, the summed percentage is 95%, and the individual percentage is 1% of the total integrated risk or risk of a specific combination of source of radioactive material, hazard, and plant operating state.
Risk-significant SSC or HFE, relative	An SSC or HFE represented by a basic event or group of basic events that contributes significantly to baseline risk. This contribution generally includes any group of basic events that has an FV importance greater than 0.005 or a RAW importance greater than 2 where the importance is normalized against the total integrated risk or risk of a specific combination of source of radioactive material, hazard, and plant operating state.

1.9.3 Absolute Risk Significance Criteria

The risk significance of items in the PRA for applications that calculate risk compared to fixed targets should be determined using absolute criteria. SRs for the selection of absolute risk criteria for such applications are included in SR RI-A2. Alternative forms of absolute risk significance criteria include frequency-consequence targets for evaluating risk significance of event sequence families and cumulative risk targets for evaluating the total integrated risk of all modeled event sequence families. Examples of cumulative targets include individual risk criteria such as those used in the Quantitative Health Objectives (QHOs) from the U.S. Nuclear Regulatory Commission (NRC) Safety Goal Policy [1-10]. Examples of criteria that may be used for this purpose are found in References [1-3] through [1-6]. This gives rise to the need to define risk significance terms that parallel those in Table 1.9-1 in this case for absolute risk significance determination. These terms are presented in Table 1.9-2.

Table 1.9-2 Absolute Risk Significance Criteria

Item	Absolute Significance Determination
Risk-significant basic event, absolute	A basic event that contributes significantly to an absolute risk significance criterion selected for RIDM. It is defined as any basic event that *(a)* contributes at least 1% to any identified absolute risk target; or *(b)* would result in exceeding the criterion if the basic event is assumed to fail with a probability of 1.0.
Risk-significant function, absolute	An event or element of a PRA model that represents the performance of a safety function that contributes significantly to a selected absolute risk criterion. It is defined as any function that either *(a)* contributes at least 1% to any identified absolute risk criterion target; or *(b)* would result in exceeding the criterion if the function is assumed to fail with a probability of 1.0.
Risk-significant contributor, absolute	Implies a significant contributor to an absolute risk significance criterion selected for RIDM. It is defined as any contributor that comprises at least 1% to any identified absolute risk target. Contributors may be defined in terms of initiating events, event sequences, event sequence families, release categories, radionuclide source, or other defined elements of the PRA model.
Risk-significant event sequence or event sequence family, absolute	An event sequence or event sequence family included in a PRA model, defined at the functional or systematic level, that makes a significant contribution to an absolute risk target selected for RIDM. It is defined as any event sequence or event sequence family that contributes at least 1% to any identified absolute risk target.
Risk-significant plant operating state, absolute	A plant operating state that is associated with one or more risk-significant event sequences families. The risk significance of a plant operating state is the total risk of the associated risk-significant event sequences and may be expressed in terms of the total integrated risk, or the risk of a specific combination of source of radioactive material, hazard, and plant operating state.
Risk-significant cutset, absolute	Implies a cutset that makes a significant contribution to any identified absolute risk target selected for RIDM. It is defined as any contributor that comprises at least 1% to any identified absolute risk target.
Risk-significant SSC or HFE, absolute	An SSC or HFE represented by a group of basic events that contributes significantly to the computed risks against a target. It is defined as any SSC that either *(a)* contributes at least 1% to any identified absolute risk target; or *(b)* would result in exceeding the risk criterion if the SSC or HFE were assumed to fail with a probability of 1.0.

1.10 SCREENING CRITERIA

This Section discusses the underlying rationale for the criteria to be used in this Standard when screening out items from being considered in constructing the PRA model. Screening is an inherent part of constructing a PRA model. It is a tool used to simplify the PRA model while retaining a clear focus on important contributors to risk. As such, the underlying screening process is to ensure that items that are screened out do not impact the integrity and insights provided by the PRA model.

Hazards, initiating events, plant operating states, sources of radioactive material, event sequences, failure modes, failure rates, plant areas, plant structures, systems, or components can each be subjected to the screening process as the PRA model is constructed. Hazard groups meeting specified screening criteria may be screened out; however, internal events and seismic events may not be screened out. However, the scope and level of detail in the PRA modeling of internal events and seismic events may be limited or simplified using acceptable screening criteria as defined in the SRs in this Standard.

The criteria to be used for screening are linked to the risk significance criteria that have been selected for the intended applications of the PRA. Risk significance criteria may vary according to the PRA application and the prevailing regulatory requirements and practices. Both absolute and relative risk significance criteria may be used to establish the risk significance of PRA model items. Technical requirements for the selection of risk significance are found under HLR-RI-A.

The basic approach to screening out items from the PRA model is to establish, using demonstrably conservative assumptions, that inclusion of the item to be screened would not result in a relative risk-significant event sequence or basic event in the PRA according to the relative risk significance criteria established for the PRA. For plants that can demonstrate that there are no risk-significant basic events or event sequences based on the selected absolute risk significance criteria, items may be screened from the PRA model when it can be shown using demonstratively conservative assumptions that the risk contribution of the item, if included, would not be risk-significant.

Table 1.10-1 specifies the general criteria (both quantitative and qualitative) that shall be used in considering whether any of the Section 1.9 items can be screened out from consideration in the construction of the PRA. These general criteria are referenced, as needed, in the individual technical elements and should be applied only as directed from the SRs in the technical element. In addition to these general criteria, the individual technical elements may also include supplemental technical element-specific criteria which should be employed in completing the screening activity.

Note that, although a hazard (or hazard group) may be screened out from being developed per the requirement of the applicable part of this Standard, the screened hazard (or hazard group) may still need to be considered in the RIDM for a specific application.

In the context of hazards that are associated with a range of severities (rather than a single discrete event), the initiating event frequency refers to the frequency with which a specified site "impact threshold" is exceeded (i.e., exceedance frequency). Impact threshold is the hazard severity at which a plant transient may occur. The manner in which these screening criteria are to be applied to each item and for each hazard is specified in the SRs of this Standard. These screening criteria are to be used only when specified in a SR. Use of alternate screening criteria to those in this Section must be justified.

Table 1.10-1 Generic Screening Criteria

Index No. SCR	Screening Type	Applicable To	Screening Criterion
SCR-1	Absolute risk contribution	Event sequence families	An event sequence family [Note (1)] subject to screening that does not exceed the selected risk significance criteria and has mean occurrence frequencies less than $1x10^{-7}$/plant-year, as estimated using a bounding or demonstrably conservative analysis.
SCR-2	Relative risk contribution	PRA model items	A PRA model item [Note (2)] subject to screening that makes less than 1% contribution to the risk of an event sequence family, as defined in the referencing SR; the total contribution of the screened items to an event sequence family may not exceed 5% of the event sequence family risk.
		Event sequence families	An event sequence family subject to screening contributes less than 0.1% of the selected risk metrics [Note (3)] as defined in the referencing SR. Limits on the total contribution of screened event sequence families are as follows: *(a)* if relative risk significance criteria are used, the total contribution of all screened out event sequence families may not exceed 5% of the selected risk metrics as defined in the referencing SR or *(b)* if absolute risk significance criteria are used, the total contribution of all screened out event sequence families may not exceed 1% of the cumulative risk targets included in the absolute risk significance criteria as defined in the referencing SR [Notes (3),(4)].
SCR-3	Qualitative evaluation	PRA model items and event sequence families	It is shown using demonstratively conservative assessments that the PRA model subject to screening does not impact the plant or is subsumed into a more frequent or more impactful event.

MANDATORY NOTES:

(1) It is noted that SCR-1 must be applied to event sequence families rather than to individually modeled event sequences so that all event sequences with a similar plant response and consequence are considered and combined according to the definition of an event sequence family. See definition of event sequence family in Section 2. This distinction is made to accommodate differences in the level of detail in modeling event sequences that are supported by this Standard. The risk significance definition selected by the users must ensure that event sequence families cannot be risk-significant below this screening frequency to ensure that risk insights are supported by the capability of PRA technology to resolve the event sequence family.

(2) PRA model items include well-defined modeling items of a PRA model including hazards; plant operating states; radionuclide sources; initiating events; event sequences; event sequence families; release categories; basic events; human failure events; and, SSCs as explained in the previous section.

(3) Selected risk metrics include all those selected by the user to represent the baseline risk for the total integrated risk, or the risk of a specific combination of radionuclide source, plant operating state, and hazard depending on what is being screened as specified in the SR that refers to these criteria. Examples include frequency of exceeding specified dose levels, individual risk of early and latent fatality, and risk metrics for other consequence measures selected for the PRA applications.

(4) Absolute risk significance criteria include frequency-consequence targets for evaluating the risk significance of individual event sequence family and cumulative risk targets. The limit on the total risk of screened event sequence families is based only on the cumulative targets.

1.11 INTERFACE WITH OTHER PRA STANDARDS

The technical requirements in this Standard were derived in part from content in ASME/ANS RA-Sb-2013 (R2019) [1-2], which was developed for currently operating LWR plants as well as other PRA trial use standards developed for LWRs by the American Nuclear Society (ANS) and the American Society of Mechanical Engineers (ASME). These trial use standards include those for LPSD PRAs [1-7], Level 2 PRA [1-8], and Level 3 PRA [1-9]. When these trial use standards are revised to meet the requirements of the American National Standards Institute, future revisions to this Standard may be necessary to incorporate applicable changes to the supporting standards. Many of the requirements from these supporting standards are applicable to non-LWRs, and other requirements were applicable with modifications to remove aspects specific to LWRs. Additional requirements were developed in this Standard to address pre-operational or bounding site PRA applications, to achieve applicability to advanced non-LWRs, and to support PRA applications unique to advanced non-LWRs. The technical approach for the derivation of these requirements is explained in Section 1.12. The major differences between this Standard and the source standards include differences in PRA scope as well as that these requirements were developed without regard to a specific reactor technology and are intended to be used for advanced non-LWR plants. A comparison of the PRAs supported by this Standard from content in ASME/ANS RA-Sb-2013 (R2019) [1-2] is provided in Table 1.11-1.

Table 1.11-1 Comparison of PRA Standards

PRA Attribute	ASME/ANS RA-Sb-2013 (R2019)	This Standard
Applicable reactor types	Current-generation LWRs	All advanced non-LWRs
Plant design, licensing, operation, life cycle stage	Currently licensed and operating plants	All stages of design, design certification, licensing, and operation
Plant operating states	Full-power plant operating state only	Risk-significant operating and shutdown plant operating states
Events and hazards	Internal hazards (e.g., internal events, internal fires and floods) external hazards (e.g., seismic events, high winds, external flooding); security threats excluded	Internal hazards (e.g., internal events, internal fires and floods) external hazards (e.g., seismic events, high winds, external flooding); security threats excluded
Success criteria basis	Prevent core damage and large early release	Prevent radioactive material release from each modeled radionuclide source
End states	Success, core damage, large early release	Success, event sequence families with similar initiating events, plant response, damage states, and release characteristics
Consequence analysis	Qualitative	Mechanistic source terms, off-site radiological consequences
Risk metrics	Core damage frequency and large early release frequency	Metrics involving frequency and consequences of modeled event sequence families [Note (1)]
Scope of event sequences covered in PRA	Beyond design-basis events involving core damage involving a single reactor	Frequent, infrequent, and rare events within and beyond design basis, including those involving two or more reactors or sources within the scope of the PRA
Sources of radioactive material considered	Reactor core	Reactor core and other sources with potential for producing risk-significant event sequences

NOTE:
(1) Selected risk metrics include all those selected by the user to represent the baseline risk for the total integrated risk, or the risk of a specific combination of radionuclide source, plant operating state, and hazard. Examples include frequency of exceeding specified dose levels, individual risk of early and latent fatality, and risk metrics for other consequence measures selected for the PRA applications.

1.12 DERIVATION OF TECHNICAL REQUIREMENTS

1.12.1 Overview

Objectives for each technical element in this Standard were developed to characterize the scope of a PRA on an advanced non-LWR nuclear power plant. The objectives from the 2013 trial use version of this Standard [1-1] were revised to achieve alignment with the supporting LWR standards [1-2, 1-7, 1-8, 1-9] and to address feedback from pilot PRAs that have been performed.

One key difference between this Standard and the supporting LWR standards is the inclusion of a "Risk Integration" technical element. This element provides requirements for integrating the results of the PRA for the definition of event sequences covering different sources of radioactive material, hazard groups, and plant operating states and the quantification of their frequencies, consequences, and uncertainties to provide an integrated assessment of risk. The need for this element stems from the approach that was adopted to provide a reactor technology inclusive PRA standard using reactor technology inclusive risk metrics. The Risk Integration element also includes the requirements for establishing the risk significance of PRA model items such as event sequences, event sequence families, basic events, and human failure events.

The technical requirements in this Standard were derived with the following inputs and considerations:

(a) For applicability to a spectrum of advanced reactor designs including gas-cooled thermal and fast reactors, LMRs, MSRs, micro-reactors, and other reactor designs that are substantially different from operating LWRs, the requirements are developed in this Standard on a reactor-technology-neutral basis.

(b) The scope of PRA elements that are included was selected based on near-term PRA applications that are

envisioned for advanced non-LWRs. Examples of the type of PRA applications that are envisioned for advanced non-LWRs include input to the selection of licensing basis; events; SSC safety classification; risk-informed evaluation of defense-in-depth; and derivation of requirements for equipment and system reliability [1-5].

(c) Existing PRA standards and draft PRA standards were reviewed for applicability of requirements to advanced non-LWR designs. Many of the reviewed requirements were found to be directly applicable, and others were modified for general applicability to any known advanced reactor PRA.

(d) New requirements were developed to address technical issues and PRA scopes not supported by the existing LWR standards.

(e) Some requirements from the 2013 trial use version of this Standard were modified to incorporate feedback from pilot PRAs that were developed and supported by that standard. This resulted in a major change in the approach to defining risk significance.

(f) Changes were made to the requirements in the 2013 trial use version of this Standard to support PRA applications using the Licensing Modernization Project (LMP) [1-5] methodology that supports the preparation of license applications for advanced non-LWRs. To support such applications, this version of this Standard includes the use of both relative and absolute risk significance metrics and criteria. Requirements for selection and application of risk significance criteria are included in the Risk Integration element. See Section 1.9 for the revised approach to risk significance and associated criteria.

1.12.2 Technical Approach to Deriving Requirements

The technical approach to deriving technical requirements for the trial use version of this Standard is discussed in Section 4.3 of Reference [1-1]. Use was made of supporting LWR PRA standards for aspects of PRA modeling independent of the reactor type. New and revised requirements were developed to make this Standard reactor technology inclusive, to expand the scope of this Standard for new applications, and to support PRAs during various plant life-cycle stages. To support stakeholder needs, the non-LWR Standard has been structured as a fully integrated standard as opposed to the multi-part format used in ASME/ANS RA-Sb-2013 (R2019) [1-2].

In modifying the requirements for this Standard, lessons learned from pilot PRAs using the trial use standard have been incorporated. Finally, it was necessary for this Standard to incorporate changes that have been made in the supporting LWR standards since issuance of the trial use standard.

1.12.3 General References for Technical Requirements

General references used to develop the technical requirements are listed here. References that are keyed to specific elements of the PRA are listed at the end of the respective section for each element in Section 4.

1.13 REFERENCES

The following is a list of publications referenced in this Standard.

[1-1] ASME/ANS RA-S-1.4-2013, "Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants," American Society of Mechanical Engineers and American Nuclear Society, 2013

[1-2] ASME/ANS RA-Sb-2013 (R2019), "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers and American Nuclear Society, 2013

[1-3] NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing," U.S. Nuclear Regulatory Commission, December 2007

[1-4] ANSI/ANS-53.1-2011 (R2016), "Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants," American Nuclear Society, 2011

[1-5] NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Advanced Reactor Licensing Basis Development," Nuclear Energy Institute, Revision 1, August 2019

[1-6] Regulatory Guide 1.233, "Guidance For a Technology-Inclusive, Risk-Informed, and Performance Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals For Non-Light-Water Reactors," U.S. Nuclear Regulatory Commission, June 2020

[1-7] ANS/ASME-58.22-2014, "Requirements for Low Power and Shutdown Probabilistic Risk Assessment," American Nuclear Society and American Society of Mechanical Engineers, 2014

[1-8] ASME/ANS RA-S-1.2-2014, "Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs)," American Society of Mechanical Engineers and American Nuclear Society, 2014

[1-9] ASME/ANS RA-S-1.3-2017, "Standard for Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications," American Society of Mechanical Engineers and American Nuclear Society, 2017

[1-10] 51 FR 28044, "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement," U.S. Nuclear Regulatory Commission, August 21, 1986

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SECTION 2 – ACRONYMS AND DEFINITIONS
CONTENTS

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SECTION 2
ACRONYMS AND DEFINITIONS

The following definitions are provided to ensure a uniform understanding of select acronyms and terms as they are specifically used in this Standard.

2.1 ACRONYMS

AC: alternating current

AEF: annual exceedance frequency

ALWR: advanced light water reactor

ANS: American Nuclear Society

AOPs: abnormal operating procedures

AOT: allowed outage time

APC: atmospheric pressure change

ASCE: American Society of Civil Engineers

ASEP: Accident Sequence Evaluation Program

ASME: The American Society of Mechanical Engineers

BOP: balance of plant

C&C: components and cladding

CC-I/CC-II: Capability Category I/Capability Category II

CCDF: complementary cumulative distribution function

CCF(s): common cause failure(s)

CCW: component cooling water

CEUS: central and eastern United States

CDF: core damage frequency

CT: cooling tower

DCF: dose conversion factor

DHR: decay heat removal

EAB: exclusion area boundary

EDG: emergency diesel generator

EOPs: emergency operating procedures

EPIX: Equipment Performance and Information Exchange System (replacement database for NPRDS)

EPRI: Electric Power Research Institute

EPZ: emergency planning zone

F-C: frequency consequence

FEG: functional equipment group

FMEA: failure modes and effects analysis

FV: Fussell-Vesely

GIP: generic implementation procedure

GMC: ground motion characterization

GRS: ground response spectra

HAZOPS: Hazards and Operability Study

HELB: high-energy line break

HEP: human error probability

HFE: human failure event

HLR: High Level Requirement

HROI: hazard range of interest

HTGR: high-temperature gas-cooled reactor

HVAC: heating, ventilation, and air conditioning

HWEL: high wind equipment list

I&C: instrumentation and control

IPEEE: individual plant examination of external events

ISI: in-service inspection

ISRS: in-structure response spectra

LBE: licensing basis event

LCO: limiting condition of operation

LERF: large early release frequency

LIP: local intense precipitation

LMP: Licensing Modernization Project

LMR: liquid metal-cooled fast reactor

LOOP: loss of off-site power (also referred to as "LOSP")

LPSD: low power and shutdown

LWR: light water reactor

MCC: motor control center

MCR: main control room

MGL: Multiple Greek Letter

MHTGR: modular high-temperature gas-cooled reactor

MOV: motor-operated valve

MSO: multiple spurious operations

MSR: molten salt reactor

NEI: Nuclear Energy Institute

NFPA: National Fire Protection Association

non-LWR: non-light water reactor

NPP: nuclear power plant

NRC: U.S. Nuclear Regulatory Commission

OSP: off-site power

P&IDs: piping and instrumentation drawings (or diagrams)

PAU: physical analysis unit

PBMR: Pebble Bed Modular Reactor

PDS: plant damage state

PGA: peak ground acceleration

PHA: process hazards analysis

PMF: probable maximum flood

PRA: probabilistic risk assessment

PRM: plant response model

PSF: performance shaping factor

QA: quality assurance

QHO: Quantitative Health Objective

RAW: risk achievement worth

RCB: reactor coolant system boundary

RCF: release category frequency

RCS: reactor coolant system

RE: reference earthquake

RES: Office of Nuclear Regulatory Research (of the NRC)

RG: Regulatory Guide (an NRC-issued communication)

RHR: residual heat removal

RIDM: risk-informed decision-making

RMW: radius of maximum winds

RPV: reactor pressure vessel

SA: spectral acceleration

SAR: safety analysis report

SBO: station blackout

SEL: seismic equipment list

SPRA: seismic probabilistic risk assessment

SR: Supporting Requirement

SRP: standard review plan

SSC(s): structure(s), system(s), and component(s)

SSEL: safe shutdown equipment list

SSHAC: Senior Seismic Hazard Analysis Committee

SSI: soil-structure interaction

SW: service water

TEDE: total effective dose equivalent

TS: Technical Specification

UHS: uniform hazard response spectrum

V/H: vertical-to-horizontal

V_L: lower bound wind speed magnitude

XFEL: external flood equipment list

2.2 DEFINITIONS

Adversely affect: to impact plant equipment items leading to equipment failure (for example, in the context of an Internal Fire PRA, a fire that includes spurious operation of devices).

Aleatory uncertainty: the uncertainty inherent in a nondeterministic (stochastic, random) phenomenon. Aleatory uncertainty is represented by modeling the phenomenon in terms of a probabilistic model. In principle, aleatory uncertainty cannot be reduced by the accumulation of more data or additional information. (Aleatory uncertainty is sometimes called “randomness.”)

Alternative assumption: one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being challenged.

As-built, as-operated: a conceptual term that represents the degree to which the PRA matches the current plant design, plant procedures, and plant performance data, relative to a specific point in time. NOTE: At the design certification stage, the plant is neither built nor operated. For these situations, the intent of the PRA model is to represent the “as-designed, as-intended-to-operate” plant.

Assumption: a judgment that is made in the development of the PRA model either for modeling convenience or because of a lack of information or state-of-knowledge. An assumption is a source of model uncertainty.

(a) An example of assumption used for modeling convenience is limiting the number of individually modeled components under the assumption that the consequence of any individual combination of components is the same.

(b) An example of assumption made for the lack of information is assuming component failure due to failure of HVAC in the absence of detailed room heat-up calculations.

At-initiator human failure event: a type of initiating event; a human failure event that causes or contributes to an initiating event (e.g., the human failure event that directly involves plant personnel actions at the time of the initiating event, including actions correctly performed but which are based on erroneous indications). This group does not include malicious acts such as sabotage. Of particular interest are those at-initiator HFEs that are directed from the main control room and in the objectives portion of Section 4.3.6 Human Reliability Analysis element.

At-power: those plant operating states characterized by the reactor being critical and producing power, with automatic actuation of critical safety systems not blocked and with essential support systems aligned in their normal power operation configuration.

Availability: the complement of unavailability.

Basic event: an event in a fault tree model that requires no further development because the appropriate limit of resolution has been reached.

Bounding site: a hypothetical site that is defined to bound the characteristics of a range of sites for use in the design of a standard plant. The site characteristics may be selected from site parameters from actual sites. For this bounding site, site-related parameters are defined using a set of scenarios that are chosen to provide appropriately high external hazard design parameter values and the most adverse meteorological conditions and population data for assessing off-site radiological impact.

Cable: referring solely to "electric cables," a construction comprising one or more insulated electrical conductors (generally copper or aluminum). A cable may or may not have other physical features such as an outer protective jacket, a protective armor (e.g., spiral wound or braided), shield wraps, and/or an uninsulated ground conductor or drain wire. Cables are used to connect points in a common electrical circuit and may be used to transmit power, control signals, indications, or instrument signals.

Cable failure mode: the behavior of an electrical cable upon fire-induced failure that may include intracable shorting, intercable shorting, and/or shorts between a conductor and an external ground. (See also *hot short*.)

Capability Category: see Table 1.4-2.

Circuit failure mode: the manner in which a conductor fault is manifested in the circuit. Circuit failure modes include loss of motive power, loss of control, loss of or false indication, open circuit conditions (e.g., a blown fuse or open circuit protective device), and spurious operation.

Cliff edge effect: an instance of a sudden large variation in plant conditions in response to a small variation in an input (e.g., change in flood height, grid perturbation based on voltage or frequency exceeding a breaker trip set point).

Coexistent hazard: a hazard that is secondary to and/or concurrent with another hazard.

Common cause failure (CCF): a failure of two or more components during a short period of time as a result of a single shared cause.

Community distribution: for any specific expert judgment, the distribution of expert judgments of the entire relevant (informed) technical community of experts knowledgeable about the given issue.

Complementary cumulative distribution function (CCDF): complement of cumulative distribution function; plot of consequence parameter being calculated against its frequency of exceedance.

Component: an item in an NPP, such as a vessel, pump, valve, or circuit breaker.

Composite variability: the composite variability includes the *aleatory* (randomness) uncertainty (β_R) and the *epistemic* (modeling and data) uncertainty (β_U). The standard deviation of composite variability β_C is expressed as $\sqrt{\beta_R^2 + \beta_U^2}$.

Concurrent hazard: a hazard that occurs simultaneously with the occurrence of another hazard as a result of a common cause (e.g., high winds concurrent with storm surge event caused by a hurricane or a moderate wind event concurrent with a large rainfall event).

Conservative: use of information (e.g., assumptions) such that the assessed outcome is meant to be less realistic in a cautious manner as compared to the expected outcome.

Cumulative distribution function: integral of the probability density function; it gives the probability of a parameter of being less than or equal to a specified value.

Damage criteria: those characteristics of the fire-induced environmental effects that will be taken as indicative of the fire-induced failure of a damage target or set of damage targets.

Damage target: see *target*.

Damage threshold: the values corresponding to the damage criteria that will be taken as indicative of the onset of fire-induced failure of a damage target or set of damage targets.

Demand-based initiating event: an initiating event that is linked to a specific activity as opposed to occurring randomly in time over the plant operating state duration. For example, in a PWR, the initiator "over-draining while reducing RCS level to mid-loop" that leads to a loss of decay heat removal would be considered a demand-based initiating event since the activity for drain down to mid-loop has been associated with historical over-draining events.

Demonstrably conservative analysis: the use of information that provides high confidence that the assessed outcome is as conservative as it is portrayed to be.

Dependency: a requirement external to an item and upon which its function depends and is associated with dependent events that are determined by, influenced by, or correlated to other events or occurrences.

Disaggregation: a technique that computes the relative contribution of an individual contributor to the total integrated risk, or risk from a specific hazard, radionuclide source, or plant operating state or key parameters (e.g., earthquake magnitude, distance) to the total integrated risk, or risk from a specific hazard, radionuclide source, or plant operating state.

Distribution system: piping, raceway, duct, or tubing that carries or conducts fluids, electricity, or signals from one point to another.

Electrical overcurrent protective device: an active or passive device designed to prevent current flow from exceeding a predetermined level by breaking the circuit when the predetermined level is exceeded (e.g., fuse or circuit breaker).

End state: the set of conditions at the end of an event sequence that characterizes the impact of the sequence on the plant or the environment. In most PRAs, end states typically include success states (i.e., those states with negligible impact) and *release categories*.

Epistemic uncertainty: the uncertainty attributable to incomplete knowledge about a phenomenon that affects our ability to model it. Epistemic uncertainty is represented by ranges of values for parameters, a range of viable models, the level of model detail, multiple expert interpretations, and statistical confidence. In principle, epistemic uncertainty can be reduced by the accumulation of additional information. (Epistemic uncertainty is typically manifested as parameter, modeling, and completeness uncertainty.)

Equipment: a term used to broadly cover the various components in an NPP. Equipment includes electrical and mechanical components (e.g., pumps, control and power switches, integrated circuit components, valves, motors, fans) and instrumentation and indication components (e.g., status indicator lights, meters, strip chart recorders, sensors). Equipment, as used in this Standard, *excludes* electrical cables.

Equipment qualification: the generation and maintenance of data and documentation to demonstrate that equipment is capable of operating under the conditions of a qualification test, or test and analysis.

Evaluator expert: an expert who is capable of evaluating the relative credibility of multiple alternative hypotheses and who is expected to evaluate all potential hypotheses and bases of inputs from proponents and resource experts, to provide both evaluator input and other experts' representation of the *community distribution*.

Event frequency: the expected number of occurrences of an event such as an *initiating event* or event sequence per unit of time, normally expressed in events per plant-operating-year (or reactor-operating-year) or events per plant-calendar-year (or reactor-calendar-year). In the context of this standard, a plant may include one or more reactors. For PRAs that are performed on multi-reactor plants, event frequencies are normally measured on a per-plant-year basis, whereas PRAs that are performed on a single reactor are normally measured on a per-reactor-year basis.

Event sequence: a representation of a scenario in terms of an *initiating event* defined for a set of initial plant conditions (characterized by a specified *plant operating state*) followed by a sequence of system, safety function, and operator failures or successes, with sequence termination with a specified end state (e.g., prevention of release of radioactive material or release in one of the reactor-specific release categories.) (See *release category*.) An event sequence may contain many unique variations of events (minimal cutsets) that are similar in how they impact the performance of safety functions along the event sequence.

Event sequence analysis: the process to determine the combinations of plant operating states, initiating events, safety functions, system failures and successes, and end states that may involve a release of radioactive material.

Event sequence family: a grouping of event sequences with similar challenges to the plant safety functions, response of the plant in the performance of each safety function, response of each radionuclide transport barrier, and end state. An event sequence family may involve a single event sequence or several event sequences grouped together. Each *release category* may include one or more event sequence families. When event sequence models are developed in great detail, identification of families of event sequences with common or similar source, initiating event and plant response facilitates application of the event sequence modeling requirements in this Standard and development of useful risk insights in the identification of risk contributors. Each event sequence family involving a release is associated with one and only one release category. (See *event sequence* and *release category*.)

Event sequence, risk-significant: see *risk-significant event sequence*.

Event tree: a logic diagram that begins with an initiating event or condition and progresses through a series of branches that represent expected system or operator performance that either succeeds or fails and arrives at either a successful or failed end state.

Event tree top event: the conditions (i.e., system behavior or operability, human actions, or phenomenological events) that are considered at each branch point in an event tree.

Expert elicitation: a formal, highly structured, and documented process whereby expert judgments, usually of multiple experts, are obtained.

Expert judgment: information provided by a technical expert, in the expert's area of expertise, based on opinion or on an interpretation based on reasoning that includes evaluations of theories, models, or experiments.

Exposed structural steel: structural steel elements that are not protected by a passive fire barrier feature (e.g., fire-retardant coating) with a minimum fire-resistance rating of 1 h.

External event: an event originating outside an NPP that directly or indirectly causes an initiating event and may cause equipment failures, operator errors, or both, that may lead to an event sequence modeled in the PRA. Events such as earthquakes, tornadoes, and floods from sources outside the plant and fires from sources outside the plant are considered external events. (See also *internal event*.) By historical convention, LOOP not caused by another external event is considered to be an internal event.

Extremely rare event: one that would not be expected to occur even once throughout the world nuclear industry over many years (e.g., <1E-6/plant-yr).

Facilitator/integrator: a single entity (individual, team, company, etc.) who is responsible for aggregating the judgments and community distributions of a panel of experts to develop the composite distribution of the informed technical community (herein called the *community distribution*).

Failure mechanism: any of the processes that result in failure modes, including chemical, electrical, mechanical, physical, thermal, and human error.

Failure mode: a specific functional manifestation of a failure (i.e., the means by which an observer can determine that a failure has occurred) by precluding the successful operation of a piece of equipment, a component, or a system (e.g., fails to start, fails to run, leaks). NOTE: In the context of Internal Fire PRA, *spurious operation* is also considered a failure mode above and beyond failures that preclude successful operation.

Failure modes and effects analysis (FMEA): a process for identifying failure modes of specific components and evaluating their effects on other components, subsystems, and systems.

Failure probability: the likelihood that an SSC will fail to operate upon demand or fail to operate for a specific mission time.

Failure rate: the expected number of failures per unit time, evaluated, for example, by the ratio of the number of failures in a population of components to the total time observed for that population.

Fault tree: a deductive logic diagram that depicts how a particular undesired event can occur as a logical combination of other undesired events.

Figure of merit: the quantitative value, obtained from a PRA analysis, used to evaluate the results of an application (e.g., RCF).

Fire analysis tool: as used in this Standard, “fire analysis tool” is broadly defined as any method used to estimate or calculate one or more physical fire effects (e.g., temperature, heat flux, time to failure of a damage target, rate of flame spread over a fuel package, heat release rate for a burning material, smoke density, etc.) based on a predefined set of input parameter values as defined by the fire scenario being analyzed. Fire analysis tools include, but are not limited to, computerized compartment fire models, closed-form analytical formulations, empirical correlations such as those provided in a handbook, and lookup tables that relate input parameters to a predicted output.

Fire area: a portion of a building or plant that is separated from other areas by rated fire barriers adequate for the fire hazard. (Note that a rated fire barrier is a fire barrier with a fire-resistance rating.)

Fire barrier: a continuous vertical or horizontal construction assembly designed and constructed to limit the spread of heat and fire and to restrict the movement of smoke.

Fire compartment[4]: a subdivision of a building or plant that is a well-defined enclosed room, not necessarily bounded by rated fire barriers. A fire compartment generally falls within a fire area and is bounded by noncombustible barriers where heat and products of combustion from a fire within the enclosure will be substantially confined. Boundaries of a fire compartment may have open equipment hatches, stairways, doorways, or unsealed penetrations. This is a term defined specifically for fire risk analysis and maps plant fire areas and/or zones, defined by the plant and based on fire protection systems design and/or operations considerations, into compartments defined by fire damage potential. For example, the control room or certain areas within the turbine building may be defined as a fire compartment.

Fire-induced initiating event: that initiating event assigned to occur in the Internal Fire Plant Response Model for a given fire scenario.

Fire modeling: as used in this Standard, “fire modeling” refers to the process of exercising a fire analysis tool including the specification and the verification of input parameter values, the performance of any required supporting calculations, the actual application of the fire analysis tool itself, and the interpretation of the fire analysis tool outputs and results.

Fire protection feature: administrative controls, fire barriers, means of egress, industrial fire brigade personnel, and other features provided for fire protection purposes.

Fire protection program: the integrated effort involving equipment, procedures, and personnel used in carrying out all activities of fire protection. It includes system and facility design, fire prevention, fire detection, annunciation, confinement, suppression, administrative controls, fire brigade organization, inspection and maintenance, training, quality assurance (QA), and testing.

Fire protection system: fire detection, notification, and fire suppression systems designed, installed, and maintained in accordance with the applicable NFPA codes and standards.

Fire-resistance rating: the time, in minutes or hours, that materials or assemblies have withstood a fire exposure as established in accordance with an approved test procedure appropriate for the structure, building material, or component under consideration.

Fire scenario: a set of elements that describes a fire event. The elements usually include a physical analysis unit, a source fire location and characteristics, detection and suppression features to be included, damage *targets*, and intervening combustibles.

[4] It is noted that the term “fire compartment” is used in other contexts, such as general fire protection engineering, and that the term’s meaning as used here may differ from that implied in an alternate context. However, the term also has a long history of use in internal fire PRA and is used in this Standard based on that history of common internal fire PRA practice.

Fire scenario selection: the process of defining a fire scenario to be analyzed in the Internal Fire PRA that will represent the behavior and consequences of fires involving one or more fire *ignition sources*. Fire scenario selection includes the identification of a fire ignition source (or set of fire ignition sources); secondary combustibles and fire spread paths; fire damage *targets*, detection and suppression systems, and features to be credited; and other factors that will influence the extent and timing of fire damage.

Fire suppression system: generally refers to permanently installed fire protection systems provided for the express purpose of suppressing fires. Fire suppression systems may be either automatically or manually actuated. However, once activated, the system should perform its design function with little or no manual intervention.

Fire wrap: a localized protective covering designed to protect cables, cable raceways, or other equipment from fire-induced damage. Fire wraps generally provide protection against thermal damage.

Flood area: an area within a plant that is defined for the purpose of performing a flooding assessment PRA. Flood areas are normally defined in terms of one or more of the following: building types; location within a building or the site; and the physical barriers that delay, restrict, or prevent the propagation of floods to adjacent areas. Flood areas refer to areas of buildings or of the site that may be flooded due to internal or external flooding sources.

Flood-induced event sequence: an event sequence that includes a *flood-induced initiating event* and the potential for undesired consequences, with a specified end state.

Flood-induced failure mechanism: the failure mechanism of an SSC induced by a flood. Possible SSC failure mechanisms include (but are not limited to) shorting out of electrical connections, blockage of air intakes, and structural damage from flood loads. In the context of external flooding, flood-induced failure mechanisms may include additional factors such as blockage of sumps (e.g., due to debris) and overtopping of barriers.

Flood-induced initiating event: an *initiating event* that is caused by a flood either directly (e.g., loss of system function caused by diversion of flow associated with the flood) or indirectly (e.g., plant shutdown caused by the loss of function of one or more flood-damaged SSCs). In the context of external flooding, flood-induced initiating events also include initiating events due to damage of SSCs from the floodwaters.

Flood initiating area (internal flooding): the area from which the flood originates.

Flood propagation path: a physical pathway that would allow the progression of a flood within and among different *flood areas.* In the context of external flooding, flood propagation paths may begin with floods that originate from a source external from the plant.

Flood rate: the flow rate of water or steam across the breach or opening in the pressure boundary of the *flood source* during the flood event. In the context of external flooding, the flood rate may also include the rate of flow of external flood water into a flood area. Depending on the context, the flood rate may be a time-dependent rate, a maximum rate, or an average rate over the duration of the flood.

Flood scenario: a description of an event that results in a flood-induced initiating event. The factors included in the definition of a flood scenario are *flood area*; *flood source*; *flood rate*; *flood propagation path*; impact on plant SSCs; human actions included in flood initiation, mitigation, and termination; and means of detection (sensors, alarms, indications, etc.).

Flood source: an inventory of water or steam normally contained within a system, tank, component, reservoir, river, lake, or ocean that provides the potential for flooding-induced failure of SSCs in the event the flood source container or pressure or retention boundary is breached.

Flood termination: as used in the definitions of *flood scenario* and *flood volume*, the cessation of the *flood rate* by isolation of the *flood source* or exhaustion of the flood source inventory.

Flood volume: the total flood volume of water released from the source from flood initiation to termination or to a specific point in time during a *flood scenario,* unless specified as the localized volume in specific flood areas for scenarios that involve multiple flood areas. Flood volume is normally used to calculate the nominal flood height, which is associated with the submergence failure cause. Water spray volumes are generally different from flood volumes, but spray water may accumulate and contribute to flood volumes.

Fragility: fragility of an SSC is the conditional probability of its failure at a given *hazard* input level. The input could be earthquake motion, wind speed, or flood level. The fragility model used in seismic PRA is known as a double lognormal model with three parameters, which are the median acceleration capacity, the logarithmic standard deviation of the *aleatory* (randomness) uncertainty in capacity, and the logarithmic standard deviation of the *epistemic* (modeling and data) uncertainty in the median capacity.

Front-line system: a system (safety or non-safety) that is capable of directly performing one of the release-mitigating functions (e.g., core heat removal, reactivity control, or reactor vessel pressure control) modeled in the PRA.

Full power: a plant operating state during which the reactor power is at or near its normal designed value. In this plant operating state, the primary system configuration (power level, pressure, temperature, boundaries, etc.) is maintained essentially constant.

Fussell-Vesely (FV): for a specified basic event, Fussell-Vesely importance is the fractional contribution to the total of a selected figure of merit (e.g., RCF) for all event sequences associated with the figure of merit containing that basic event. For PRA quantification methods that include non-minimal cutsets and success probabilities, the Fussell-Vesely importance measure is calculated by determining the fractional reduction in the total figure of merit brought about by setting the probability of the basic event to zero.

Ground acceleration: acceleration at the ground surface produced by seismic waves, typically expressed in units of g, the acceleration of gravity at the earth's surface.

Harsh environment: an abnormal, severe environment (e.g., high or low temperature, humidity, corrosive conditions) expected as a result of the event sequences modeled in the PRA.

Hazard: a phenomenon that challenges the safe operation of a facility. A hazard is a subset of a *hazard group* and a superset of *hazard events*. Hazards in the internal events hazard group include RCB breaches and LOOPs. In some cases, a hazard group may consist only of one hazard (i.e., the seismic hazard), in which case the hazard and hazard group are considered to be synonymous.

Hazard analysis: the process to determine an estimate of the expected frequency of exceedance (over some specified time interval) of various levels of some characteristic measure of the intensity of a hazard (e.g., PGA to characterize ground shaking from an earthquake). The time period of interest is typically 1 yr, in which case the estimate is called the annual frequency of exceedance.

Hazard event: an event brought about by the occurrence of the specified *hazard*. A hazard event is described in terms of the specific levels of severity of impact that a hazard can have on the plant. For example, an internal flood event would be expressed in terms of the specific flood source and its local impact, such as the resulting water levels in affected plant areas or the extent of the area subjected to spray; a seismic event would be expressed in terms of *spectral acceleration* and associated spectral shape; a transient event would be expressed in terms of the plant systems affected by the event.

Hazard group: a group of *hazard*s that result in similar effects on or challenges to a facility. A hazard group is a subset of a hazard type and a superset of hazards. The hazards in a given hazard group may be assessed using a common approach, methods, and likelihood data for characterizing the effect on the plant. Examples of hazard groups include internal events, internal floods, seismic events, and high wind events. In some cases, a hazard group may consist only of one hazard (i.e., the seismic hazard), in which case the hazard group and hazard are considered synonymous.

Hazard, intrinsic: a hazard that is inherent to the reactor's safety design approach and is considered in the development of event sequence models, quantification of event sequence frequencies, and development of mechanistic source terms. Intrinsic hazards include chemical and physical interactions that are associated with the inherent features of the reactor. Examples include core and graphite oxidation associated with air or water ingress into an HTGR reactor coolant system, sodium water reactions in sodium cooled reactors, and adverse chemical reactions in molten salt reactors. The treatment of intrinsic hazards is within the scope of the internal events PRA model.

Hazard type: a hazard type is a superset of hazard groups. Internal hazards include hazard groups such as internal events, and internal fire and external hazards include hazard groups such as the seismic hazard and external flooding.

High-energy arcing fault: an electrical arc that leads to a rapid release of electrical energy in the form of heat, vaporized copper, and mechanical force.

High-energy line: a pipe or piping system component is classified as high energy if it contains water or steam at maximum operating temperature exceeding 200°F or maximum operating pressure exceeding 275 psig.

High-energy line break (HELB): a break or breach in a high-energy line.

High-hazard fire source: a fire source that can lead to fires of a particularly severe and challenging nature. High-hazard fire sources would include, but are not limited to, the following: catastrophic failure of an oil-filled transformer, an unconfined release of flammable or combustible liquid, leaks from a pressurized system containing flammable or combustible liquids, and risk-significant releases or leakage of hydrogen or other flammable gases.

High winds: tornadoes, hurricanes (or cyclones or typhoons as they are known outside the United States), extratropical (thunderstorm) winds, and other wind phenomena depending on the site location.

Hot short: individual conductors of the same or different cables coming in contact with each other where at least one of the conductors involved in the shorting is energized, resulting in an impressed voltage or current on the circuit being analyzed.

Human error: any human action that exceeds some limit of acceptability, including inaction where required, excluding malevolent behavior.

Human error probability (HEP): a measure of the likelihood that plant personnel will fail to initiate the correct, required, or specified action or response in a given situation, or by commission performs the wrong action. The HEP is the probability of the HFE.

Human failure event (HFE): a basic event that represents a failure or unavailability of a component, system, or function that is caused by human inaction, or an inappropriate action.

Human reliability analysis: a structured approach used to identify potential HFEs and to systematically estimate the probability of those events using data, models, or expert judgment.

Human response action: a post-initiator operator action, following a cue or symptom of an event, taken to satisfy the procedural requirements for control of a function or system.

Ignition frequency: frequency of fire occurrence generally expressed as fire ignitions per plant-year.

Ignition source: a piece of equipment or an activity that causes fire.

Initiating event: a perturbation to the plant during a plant operating state that challenges plant control and safety systems whose failure could potentially lead to an undesirable end state and/or radioactive material release. An initiating event is defined in terms of the change in plant status that results in a condition requiring a response to mitigate the event or to limit the extent of plant damage caused by the initiating event. An initiating event may result from human causes, equipment failure from causes internal to the plant (e.g., hardware faults, flood, or fires) or external to the plant (e.g., earthquakes or high winds), or combinations thereof.

Initiator: see *initiating event*.

Insights: information that provides an understanding and explanation of what is and is not important to the analysis.

Integrator: a single entity (individual, team, company, etc.) who is ultimately responsible for developing the composite representation of the informed technical community (herein called the *community distribution*). This sometimes involves informal methods, such as deriving information relevant to an issue from the open literature or through informal discussions with experts, and sometimes involves more formal methods.

Intensity: a measure of the impact of a *hazard.*

Intercable (as in "intercable conductor-to-conductor short circuit"): electrical interactions (shorting) between the conductors of two (or more) separate electrical cables. (See also *intracable.*)

Intermediate state: the grouping of event sequences at some point short of the final end states as a way of organizing the PRA, typically to aid in developing intermediate results (e.g., fuel damage, PDSs). Sometimes referred to in PRA modeling as "pinch point."

Internal event: a hazard group that encompasses events other than floods or fires that result from or involve mechanical, electrical, structural, or human failures from causes originating within a nuclear power plant or losses of off-site power (except when caused by another hazard) that directly or indirectly cause an *initiating event* and may cause safety system failures or operator errors that may lead to a release of radioactive materials.

Internal fire PRA plant response model: a representation of a combination of equipment, cable, circuit, system, function, and operator failures or successes of an event sequence that when combined with a fire-induced initiating event can lead to undesired consequences with a specified end state (e.g., release category).[5]

Intracable (as in "intracable conductor-to-conductor short circuit"): electrical interactions (shorting) between the conductors of one multiconductor electrical cable. (See also *intercable*.)

Key assumption: an assumption made in response to a key source of uncertainty in the knowledge that an alternative assumption would produce different results, or an assumption that results in an approximation made for modeling convenience in the knowledge that a more detailed model would produce different results. The term "different results" refers to a change in the plant risk profile (e.g., RCF, the set of initiating events, and event sequences that contribute most to the RCF) and the associated changes in insights derived from the changes in risk profile.

Key safety functions: the minimum set of safety functions that must be maintained to prevent a radioactive material release. These include reactivity control, preservation of barriers to release, and DHR in appropriate combinations to prevent a large release. The specific set of safety functions necessary to prevent each release category is reactor specific.

Key source of uncertainty: a source of uncertainty is considered to be key to a risk-informed decision when it could impact the PRA results that are being used in a decision and, consequently, may influence the decision being made. An impact on the PRA results could include the introduction of a new functional event sequence or other changes to the risk profile (e.g., total integrated risk, risk of a source, plant operating state, and hazard group, frequency of an event sequence or event sequence family, importance measures). Key sources of uncertainty are identified in the context of an application (note that for certain applications, the base model is used).

Level of detail: the degree to which (i.e., amount of) information is discretized and included in the model or analysis.

Licensee-controlled area: areas of the plant site that are directly controlled by the NPP licensee.

Low power: a plant operating state (or set of plant operating states) during which the reactor is at reduced power, below nominal full-power conditions. In these plant operating states, the power level may be changing as the reactor is shutting down, starting up, or transitioning to a new power level required by a *plant evolution* (e.g., online refueling, online maintenance, etc.), or the power level may be constant at a reduced level. The power level that distinguishes nominal full power from low power is the power level below which there may be a significant increase in the likelihood of a plant trip.

[5] This definition has been adapted to suit internal fire analysis needs from the definition of "event sequence." A variety of equivalent terms has been used in other fire PRA-related documents including, but not limited to, post-fire safe shutdown model, internal fire PRA model, and post-fire plant response model.

Master logic diagram: summary fault tree constructed to guide the identification and grouping of initiating events and their associated sequences to ensure completeness.

May: used to state an option to be implemented at the user's discretion.

MCR-directed at-initiating event activity: a planned interaction directed by the operators in the main control room (MCR) or other command and control location, such as to realign the plant operating configuration or to change the plant operating parameters, which leads to an *initiating event* (at-initiator), and the same operators will also direct the post-initiating event response. Thus, there is the potential for a human reliability dependency between the *initiating event* and the plant response since both are directed by operators from a controlling station (typically the MCR).

Mechanistic source term: see *source term, mechanistic*.

Method: an analytical approach used to satisfy a Supporting Requirement or collection thereof in the PRA. An analytical approach is generally a compilation of the analyses, tools, assumptions, and data used to develop a model.

Mission time: the time period that a system or component is required to operate to successfully perform its function.

Mitigating structure, system, and component: an SSC that performs a function to mitigate the consequences of an event such as by protecting a *radionuclide transport barrier*, performing a safety function, or limiting or preventing a release of radioactive material from a source.

Mode: status of plant operation, as defined by plant Technical Specifications.

Model: a qualitative or quantitative representation that is constructed to portray the inherent characteristics and properties of what is being represented (e.g., a system, component or human performance, theory or phenomenon). A model may be in the form, for example, of a structure, schematic, or equation. Method(s) are used to construct the model under consideration.

Multi-compartment fire scenario: a fire scenario involving targets in a room or fire compartment other than, or in addition to, the one where the fire was originated.

Multiple spurious operations (MSO): concurrent spurious operations of two or more equipment items.

Mutually exclusive events: a set of events where the occurrence of any one precludes the simultaneous occurrence of any remaining events in the set.

Newly developed method: a method used in a PRA that has either been developed separately from a state-of-practice method or is one that involves a fundamental change to a state-of-practice method. A newly developed method is not a state-of-practice or a consensus method.

Nominal full power: see *full power*.

Nonsuppression probability: the probability of failing to suppress a fire before target damage occurs.

Operating time: total time during which components or systems are performing their designed function.

Outage: the entire set of plant operating states with the plant subcritical. This term is used interchangeably with the term "shutdown" (see discussion under *shutdown*).

Outage types: the term used to describe the general cause of the plant being subcritical. Different outage types result from maintenance and refueling requirements that necessitate different LPSD evolutions and resulting plant operating states. For example, a *refueling outage* type may involve fuel movement operations, whereas a maintenance outage conducted to repair piping would be a different outage type.

Peak ground acceleration (PGA): the maximum absolute value of acceleration displayed on an accelerogram; the largest ground acceleration produced by an earthquake at a site.

Per calendar-year: units for the frequency of an *initiating event*, *event sequence*, or *release category*, the calculation of which includes contributions from each plant operating state, taking into account the fraction of time spent in that plant operating state, normalized to 1 calendar-yr. Thus, the results from each plant operating state per calendar-year can be summed to give the total quantitative risk results. Also, note that the total risk per calendar-year does not represent any actual year of operation since it includes all possible plant operating states, some of which may occur only in outages that are less frequent than yearly.

Performance shaping factor (PSF): a factor that influences *human error probabilities* as considered in a PRA's Human Reliability Analysis and includes such items as level of training, quality/availability of procedural guidance, time available to perform an action, etc.

Physical analysis units (PAU): the spatial subdivisions of the plant upon which an internal flood PRA or an internal fire PRA is based. The physical analysis units are generally defined in terms of flood or fire areas or flood or *fire compartments* under the plant partitioning technical element.

Plant: a general term used to refer to a nuclear power facility (for example, "plant" could be used to refer to a single reactor or multi-reactor site).

Plant boundary: defined by the user based on the scope of plant structures.

Plant configuration: plant conditions including mode, reactor power and decay heat level, RCS conditions (e.g., temperature, pressure), RCS status (e.g., pressure boundary open or closed), reactor building status, fire and flood barrier status, equipment alignment (e.g., number of pumps operating, number of pumps in standby), and equipment in service or out of service for test and maintenance.

Plant critical year: a calendar year in the operating life of a plant, assuming that the plant operated continuously for a year.

Plant damage state (PDS): group of event sequences identifying intermediate states that have similar characteristics with respect to event sequence progression, barrier response or mitigation system operability.

Plant evolution: a series of connected or related activities where the plant transitions from one plant operating state to another (e.g., a transition from full-power to low-power level or shutdown), or changes to the plant conditions with various combinations of equipment out of service for maintenance. Not all plant evolutions involve a change in reactor power. Plant evolutions may be modeled as a series of plant operating states. Outage types are subtypes of a plant evolution, though not all plant evolutions involve an outage. A refueling outage is a specific example of a plant evolution. Reducing power to 30% to conduct maintenance or an operational activity is another example of a low-power evolution. Plant evolutions may be described by a transition down to the plant operating state where the activity is conducted, followed by a transition back to full power.

Plant operating state: a standard arrangement of the plant during which the plant conditions are relatively constant, are modeled as constant, and are distinct from other configurations in ways that impact risk. Plant operating state is a basic modeling device used for a phased-mission risk assessment that discretizes the plant conditions for specific phases of plant evolution. Examples of such plant conditions include core decay heat level, reactor coolant level, coolant temperature, coolant vent status, reactor building status, and DHR mechanisms. Examples of risk impacts that are dependent on the plant operating state definition include the selection of initiating events, initiating event frequencies, definition of event sequences, success criteria, and event sequence quantification.

Plant-operating-state-year: an equivalent calendar-year of operation of a plant in a particular plant operating state.

Plant response model: a logic model, including the event trees and fault trees and the various SSC and human failures, that is used to delineate and evaluate the modeled event sequences conditional on the occurrence of a hazard event (or hazard group).

Plant-specific data: data consisting of observed sample data from the plant being analyzed.

Plant-year: a calendar-year in the operating life of a plant, regardless of power level.

Point estimate: an estimate of a parameter in the form of a single number.

Post-initiator human failure events: human failure events that represent the impact of human errors committed during response to abnormal plant conditions.

Pre-initiator human failure events: human failure events that represent the impact of human errors committed during actions performed prior to the initiation of an event (e.g., during maintenance or the use of calibration procedures).

Primary hazard: those hazards that are not the consequence of other preceding hazards.

Prior distribution (priors): in Bayesian analysis, the expression of an analyst's prior belief about the value of a parameter prior to obtaining sample data.

Probabilistic risk assessment (PRA): a quantitative assessment of the risk associated with plant operation and maintenance that is measured in terms of frequency of occurrence and consequences of *event sequences*, *event sequence families*, or *release categories* [also referred to as a probabilistic safety analysis (PSA)].

Probabilistic risk assessment (PRA) application: a documented analysis based in part or whole on a plant-specific PRA that is used to assist in decision-making regarding the design, licensing, procurement, construction, operation, or maintenance of an NPP.

PRA maintenance: a change in the PRA that does not meet the definition of PRA upgrade.

PRA upgrade: a change in the PRA that results in the applicability of one or more SRs or Capability Categories (e.g., the addition of a new hazard model) that were not previously assessed in a peer review of the PRA, an implementation of a PRA method in a different context, or the incorporation of a method not previously used.

Probability of exceedance (as used in seismic hazard analysis): the probability that a specified level of ground motion for at least one earthquake will be exceeded at a site or in a region during a specified exposure time.

Quantitative Health Objective(s) [QHO(s)]: U.S. Nuclear Regulatory Commission term for numerical criteria for the acceptable levels of risk to public health and safety in the population surrounding NPPs that satisfy NRC's reactor safety goals. These QHOs are expressed in terms of the annual average individual probability of death due to acute radiation syndrome within 1 mile of the site boundary and the annual average individual probability of death due to latent cancer per year within 10 miles of the site boundary of an NPP, and these are set at <0.1% of the levels due to nonnuclear causes.

Raceway: an enclosed channel of metal or nonmetallic materials designed expressly for holding wires, cables, or bus bars, with additional functions as permitted by code. Raceways include, but are not limited to, rigid metal conduit, rigid nonmetallic conduit, intermediate metal conduit, liquid-tight flexible conduit, flexible metallic tubing, flexible metal conduit, electrical nonmetallic tubing, electrical metallic tubing, underfloor raceways, cellular concrete floor raceways, cellular metal floor raceways, surface raceways, wireways, and busways.

Radiation dose: energy deposited into human tissue from external, inhaled, or ingested sources of alpha, beta, or gamma radiation, measured in units of rem or sieverts (1 Sv = 100 rem).

Radionuclide group: a set of radionuclides that are treated as a single representative species for the purposes of calculating release from fuel and transport to the environment. Physical and transport properties for the single representative species are assumed to apply to all other radionuclides within the group. The group is usually composed of all nuclides of a common element and all nuclides of other elements that have similar physical and chemical properties.

Radionuclide release category: See *release category*.

Radionuclide transport barrier: a passive SSC that is designed to retain radionuclides and/or to mitigate the radionuclide release source term during an event sequence. Such barriers include the physical barriers such as the fuel barrier, RCB, and reactor building. Barriers also include the time delays for radioactive decay, deposition, and revaporization of the released material within and between the respective physical barriers.

Radionuclide transport barrier bypass: a direct or indirect flow path that may allow the release of radioactive material from the RCS directly to the environment bypassing one or more *radionuclide transport barriers*.

Radionuclide transport barrier challenge: the challenge to the integrity of one or more *radionuclide transport barriers* by severe event sequence conditions (e.g., plant thermal hydraulic conditions or phenomena) during an event sequence.

Radionuclide transport barrier failure: loss of integrity of a *radionuclide transport barrier* to perform its safety functions in the mitigation of a release of radionuclides to the environment.

Radionuclide transport barrier failure mode: the manner in which a radionuclide release pathway is created by a *radionuclide transportation barrier failure* (e.g., structural failures, isolation failures, barrier bypass events, and human-induced failures).

Radionuclide transport barrier performance: a measure of the response of a *radionuclide transport barrier* to event sequence conditions.

Randomness (as used in seismic fragility analysis): the variability in seismic capacity arising from the randomness of the earthquake characteristics for the same acceleration and to the structural response parameters that relate to these characteristics.

Rare event: one that might be expected to occur only a few times throughout the world nuclear industry over many years (e.g., <1E-4/plant-yr).

Reactor coolant system boundary (RCB) breach: a breach in the RCB that would result in a loss of RCS pressure and/or loss of coolant inventory. In LWRs, RCB breaches are referred to as Loss-of-Coolant Accidents.

Reactor coolant system boundary (RCB) breach, excessive: a breach in the RCB that is beyond the capacity of the mitigating systems either due to its size or location.

Realism: an accurate representation (to the extent practical) of the expected response of the as-built, as-operated plant or as-designed plant.

Recovery: restoration of a function lost as a result of a failed SSC by overcoming or compensating for its failure. It is generally modeled by using Human Reliability Analysis techniques.

Refueling outage: an outage type that occurs on a periodic basis, during which a portion of the spent nuclear fuel is replaced with new (unburned) fuel.

Release category: the grouping of one or more event sequences or event sequence families based on common or similar *mechanistic source terms*. In this context, "similar" depends on the level of fidelity of the analysis and the number of release categories used to span the entire spectrum of possibilities within the scope of the PRA model. Similarity is generally measured in terms of the overall (cumulative) release of activity to the environment, the time at which the release begins, and (in certain applications) other physical characteristics of the source term.

Release category frequency: the expected number of occurrences of a specified *release category* per unit time.

Reliability: the complement of unreliability.

Repair: restoration of a failed SSC by correcting the cause of failure and returning the failed SSC to its modeled functionality. Generally modeled by using actuarial data.

Repair time: the period from identification of a component failure until it is returned to service.

Resource expert: a technical expert with knowledge of a particular technical area of a PRA.

Response: a reaction to a cue for action in initiating or recovering a desired function.

Response models: representation of post-initiator control room operator actions, following a cue or symptom of an event, to satisfy the procedural requirements for control of a function or system.

Response spectrum: a curve calculated from an earthquake accelerogram that gives the value of peak response in terms of acceleration, velocity, or displacement of a damped linear oscillator (with a given damping ratio) as a function of its period (or frequency).

Risk: frequency and consequences of an event, as expressed by the "risk triplet" that is the answer to the following three questions: (a) What can go wrong? (b) How likely is it? and (c) What are the consequences if it occurs? In this Standard, Question (a) is answered by the definition of event sequences, event sequence families, and release categories; Question (b) by estimating the frequency of event sequences on a per plant year basis where a plant may be comprised of one or more reactors and non-reactor radionuclide sources; Question (c) is quantified using radionuclide consequence metrics such as site boundary dose, population dose, early or latent health effects, or individual or societal risk.

Risk achievement worth (RAW) importance measure: for a specified basic event, risk achievement worth importance represents the increase in a selected figure of merit when an SSC is assumed to be unable to perform its function due to testing, maintenance, or failure. It is the ratio or interval of the figure of merit, evaluated with the SSC's basic event probability set to one, to the base case figure of merit.

Risk metrics: risk is defined in terms of the frequency of a given level of consequence. The risk metrics in this Standard include the product of the mean frequency and mean consequence of an event sequence, the sum of the product of mean frequencies and consequences over a group of event sequences, and the frequency of exceeding a given level of consequence for an event sequence or group of event sequences. In aggregating the risk over a group of event sequences, the risk may be associated with an event sequence family, a release category, a specific combination of plant operating state, source of radioactive material, and hazard group, or the total integrated risk of the plant. Frequencies of event sequences are expressed in terms of events per plant-calendar-year where a plant may include two or more reactors and radionuclides sources from shared facilities. Consequences are expressed in terms of off-site radiological metrics such as site boundary dose, population dose, early and latent health effects, and individual risks.

Risk-relevant damage targets: any equipment item or cable whose operation is credited in the Internal Fire Plant Response Model or whose operation may be required to support a credited postfire operator action.

Risk-relevant ignition source: any ignition source included in the internal fire PRA fire scenario definitions that could cause a fire that might induce a plant initiating event or adversely affect one or more damage targets.

Risk-significant basic event, absolute: see Table 1.9-2.

Risk-significant basic event, relative: see Table 1.9-1.

Risk-significant contributor, absolute: see Table 1.9-2.

Risk-significant contributor, relative: see Table 1.9-1.

Risk-significant cutset, absolute: see Table 1.9-2.

Risk-significant cutset, relative: see Table 1.9-1.

Risk-significant event sequence or event sequence family, absolute: see Table 1.9-2.

Risk-significant event sequence or event sequence family, relative: see Table 1.9-1.

Risk-significant plant operating state, absolute: see Table 1.9-2.

Risk-significant plant operating state, relative: see Table 1.9-1.

Risk-significant SSC or human failure event, absolute: see Table 1.9-2.

Risk-significant SSC or human failure event, relative: see Table 1.9-1.

Safe stable state: a plant condition, following an initiating event, in which plant conditions are controllable at or near desired values.

Safety function: a function that must be performed to control the sources of energy and radiation hazards in the plant and to maintain the integrity or mitigate the releases from one or more radionuclide transport barriers.

Screening: a process that eliminates items from further consideration based on their negligible contribution to the probability of an accident or its consequences, or from further analysis of a specific issue.

Screening criteria: the values and conditions used to determine whether an item is a negligible contributor to the probability of an event sequence or its consequences, or from further analysis of a specific issue.

Secondary combustible: combustible or flammable materials that are not a part of the fire ignition source that may be ignited if there is fire spread beyond the fire ignition source.

Secondary hazard: used in connection with, and contrast to, a primary hazard. It is an additional hazard effect that is induced by the primary hazard.

Seismic margin: expressed in terms of the earthquake motion level that compromises plant safety, specifically leading to an undesired end state. The margin concept can also be extended to any particular structure, function, system, equipment item, or component for which "compromising safety" means sufficient loss of safety function to contribute to an undesired end state either independently or in combination with other failures.

Seismic source: a general term referring to both seismogenic sources and capable tectonic sources. A seismogenic source is a portion of the earth assumed to have a uniform earthquake potential (same expected maximum earthquake and recurrence frequency), distinct from the seismicity of the surrounding regions. A capable tectonic source is a tectonic structure that can generate both vibratory ground motion and tectonic surface deformation such as faulting or folding at or near the earth's surface. In a probabilistic seismic hazard analysis, all seismic sources in the site region with a potential to contribute to the frequency of ground motions (i.e., the hazard) are included.

Seismic spatial interaction: an interaction that could cause an equipment item to fail to perform its intended safety function. It is the physical interaction of a structure, pipe, distribution system, or other equipment item with a nearby item of safety equipment caused by relative motions from an earthquake. The interactions of concern are (a) proximity effects, (b) structural failure and falling, and (c) flexibility of attached lines and cables.

Severity factor: the probability that fire ignition would include certain specific conditions that influence its rate of growth, level of energy emanated, and duration (time to self-extinguishment) to levels at which target damage is generated.

Shall: used to state a mandatory requirement.

Should: used to state a recommendation.

Shutdown: the collection of plant operating states during which the reactor is subcritical. This term is interchangeable with the term "outage."

Skill of the craft: the level of skill expected of the personnel performing the associated function.

Source of model uncertainty: a source related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, or introduction of a new initiating event). A source of model uncertainty is labeled "key" when it could impact the PRA results that are being used in a decision, and consequently may influence the decision being made. Therefore, a key source of model uncertainty is identified in the context of an application. This impact would need to be significant enough that it changes the degree to which the risk acceptance criteria are met, and therefore could potentially influence the decision.

Source term: see *source term, mechanistic*.

Source term, mechanistic: the characteristics of a radionuclide release at a particular location, including the physical and chemical properties of released material, release magnitude, heat content (or energy) of the carrier fluid, and location relative to local obstacles that would affect transport away from the release point and the temporal variations in these parameters (e.g., time of release duration, etc.) that are calculated using models and supporting scientific data that simulate the physical and chemical processes that describe the radionuclide inventories and the time-dependent radionuclide transport mechanisms that are necessary and sufficient to predict the source term.

Spectral acceleration: given as a function of period or frequency and damping ratio (typically 5%), it is equal to the peak relative displacement of a linear oscillator of frequency f attached to the ground times the quantity $(2\pi f)^2$. It is expressed in gravitational acceleration (g) or centimeters per second squared (cm/s^2).

Split fraction: a unitless quantity that represents the conditional (on preceding events) probability of choosing one direction rather than the other through a branch point of an event tree.

Spurious operation: the undesired operation of equipment resulting from a fire that could affect the capability to achieve and maintain safe shutdown.

Standby system: a system that is not normally operating but is intended to be ready to operate upon demand.

Startup: a plant operating state during which the reactor power level is increased from low power to full power following a plant outage.

State of practice: those practices that are widely accepted and implemented throughout the nuclear industry have been shown to be technically acceptable in documented analyses or engineering assessments and have been shown to be acceptable in the context of the intended application.

State-of-knowledge correlation: the correlation that arises between sample values when performing uncertainty analysis for cutsets consisting of basic events using a sampling approach (such as the Monte Carlo method); when included, this results, for each sample, in the same value being used for all basic event probabilities to which the same data applies.

Station blackout: complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant.

Statistical model: a model in which a modeling parameter or behavior is treated as a random variable with specified statistical characteristics.

Success criteria: criteria for establishing the minimum number or combinations of systems or components required to operate, operator actions, or minimum levels of performance per component during a specific period of time, to ensure that the safety functions are satisfied.

Support system: a system that provides a support function (e.g., electric power, control power, or cooling) for one or more other systems.

System failure: the loss of the ability of a system to perform a modeled function.

Target: may refer to a high wind, fire damage target and/or to an ignition target. A fire damage target is any item whose function can be adversely affected by the modeled fire. Typically, a fire damage target is a cable or equipment item that belongs to the Internal Fire PRA cable or equipment list and that is included in event trees and fault trees for fire risk estimation. An ignition target would be any flammable or combustible material to which fire might spread.

Target set: a group of damage targets that will be assumed to suffer fire-induced damage based on the same damage criteria and damage threshold in any given fire scenario. The collection of target sets associated with a fire scenario often represents a subset of the damage targets present in the fire compartment but may also encompass all risk-relevant damage targets in a single physical analysis unit or a collection of damage targets in multiple physical analysis units. This definition implies that all members of any single target set will be assumed to fail when the first member of the target set fails (i.e., damage based on the same damage criteria and damage threshold). Progressive or time-dependent states of fire damage may be represented through the definition of multiple target sets for a single fire scenario (e.g., cables in raceways directly above a fire source versus cables in raceways remote from the fire source). The level of detail associated with target set definition will generally parallel the level of detail employed in fire scenario selection and analysis (e.g., screening level analysis versus detailed analysis).

Technology-specific data: data consisting of observed sample data from experiments that were conducted to support the technology development of the plant being analyzed.

Time available: the time period from the presentation of a cue for human action or equipment response to the time of adverse consequences if no action is taken.

Time required: the time needed by operators to successfully perform and complete a human action.

Top event: the undesired state of a system in the fault tree model (e.g., the failure of the system to accomplish its function) that is the starting point (at the top) of the fault tree.

Transition: A change in plant configuration, for example, a change in plant configuration to prepare for refueling.

Truncation limit: the numerical cutoff value of probability or frequency below whose results are not retained in the quantitative PRA model or used in subsequent calculations (such limits can apply to initiating event frequencies, event sequences/cutsets, system-level cutsets, and sequence/cutset database retention).

Unavailability: the probability that a structure, system, or component (SSC) is not capable of supporting its function including, but not limited to, the time it is disabled for test or maintenance. Unavailability is one aspect of an SSC's failure to perform its function and is distinct from unreliability. Total system unavailability includes unreliability.

Uncertainty: a representation of the confidence in the information or state-of-knowledge about the parameter values and models used in constructing the PRA.

Uncertainty analysis: the process of identifying and characterizing the sources of uncertainty in the analysis and evaluating their impact on the PRA results and developing a quantitative measure to the extent practical.

Uniform hazard response spectrum (UHS): a plot of a ground response parameter (for example, spectral acceleration or spectral velocity) that has an equal likelihood of exceedance at different frequencies.

Unreliability: the probability that an SSC will not perform its specified function under given conditions upon demand or for a prescribed time. Unreliability is one aspect of an SSC's failure to perform its function and is distinct from unavailability.

Walkdown: the physical inspection of relevant areas of the NPP site (and its surroundings, as necessary) to obtain or confirm information such that the PRA model represents the as-built and as-operated plant.

Warning time: elapsed time from the order to evacuate until the start of the release. For external flood analysis, the warning time is a lead time for a flood arrival. The weather forecast and the predicted flood wave traveling time or the rate of rise of the flood waters should be determined to establish the available lead time for a plant responding to an external flood.

2.3 REFERENCE

The following is the publication referenced in this Standard.

[2-1] NUREG/CR-1278, SAND80-0200, A. D. SWAIN and H. E. GUTTMANN, "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications," U.S. Nuclear Regulatory Commission/ Sandia National Laboratories, August 1983

SECTION 3 – RISK ASSESSMENT APPLICATION PROCESS
CONTENTS

(The text presented in blue font in this standard comprise hyperlinks to enable efficient access to referenced sections and elements, equirements, notes, references, etc.)

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SECTION 3
RISK ASSESSMENT APPLICATION PROCESS

3.1 OVERVIEW OF APPLICATION PROCESS

This Section describes required activities to establish the capability of a PRA to support a particular risk-informed application. For this Section, "PRA" (or "PRA model") can refer to either an integrated model that includes all relevant sources of radioactive material, plant operating states, and hazard groups or multiple PRA models that address one or more parts of the PRA. For a specific application, PRA capabilities are evaluated in terms of Capability Categories for individual Supporting Requirements (SRs) rather than by specifying a single Capability Category for the whole PRA. Depending on the application, the required PRA capabilities may vary over and within different parts of the PRA model and the PRA elements in this Standard. In addition, for some applications, only a subset of the technical requirements may apply. The process includes consideration of whether a peer review that meets the requirements for peer review defined in Section 6 is necessary or not.

The technical requirements for performance of PRAs presented in Section 4 are intended for a wide range of applications and purposes over the plant design and operation life cycle stages. Application examples include support of design decisions, creation of a general-purpose risk tool that will evolve as the design matures, and risk-informing the decision-making process on specific technical issues. The application process in this Standard has been designed to be used during all life cycle stages, including design, licensing, and pre-operation stages, as well as operation and decommissioning, etc.

Figure 3-1 shows one logical ordering for the process. However, although the specified activities are required, their order of execution may vary. As shown in the dashed-line boxes, there are five stages to the process:

(a) Stage A. The plant life cycle stage is characterized as well as the PRA application or applications to be supported in this stage. Next, the site characteristics are defined based on the plant life cycle stage. Prior to site selection, the PRA may be for a bounding site such as the case for a design certification application. The scope and level of detail of the PRA is then selected based on the design and site characteristics and the state of design and site information available to support the PRA. The PRA scope may include the selection of hazard groups, plant operating states, and extent of resolution of the consequences of the PRA event sequences. Then, depending on the type of application, the risk significance criteria appropriate for the application are selected. For applications involving the need for a calculation of baseline risk, relative risk significance criteria are selected. Applications that involve comparisons of the PRA results against fixed risk targets will need the use of absolute risk significance criteria. See Section 1.9 for more discussion of risk significance criteria.

An application is defined in terms of the structures, systems, and components (SSCs) and activities affected by the proposed activity. For the particular application, the affected portions of a PRA are determined (i.e., the relevant portions), and the radionuclide sources, plant operating states, and hazard group(s) needed to be addressed in the application, the scope within the PRA related to the application, and risk metrics needed to support the application are identified. By using an understanding of the cause-and-effect relationship between the application and the portions of a PRA model that are particularly sensitive to the proposed change, the risk significance of each portion of the PRA necessary to support the application is determined. The SRs relevant to the different portions of a PRA within the scope, across the elements and possibly within each element, may be required to have different Capability Categories to support the application, and some portions of a PRA may be irrelevant to the application. Whether the application can be supported using conservative assumptions or whether realistic risk estimates are needed factors into the selection between Capability Category I (CC-I) or Capability Category II (CC-II).

(b) Stage B. The relevant portions of the PRA are examined to determine whether the PRA scope and level of detail and the risk metrics calculated by the PRA are sufficient for the application. Portions of the PRA may be defined as different sources of radioactive material, specific hazard groups, plant operating states, hazard groups, initiating events, event sequences, and/or release categories. If the relevant portions of the PRA or the calculated risk metrics are found lacking in one or more areas, they may be upgraded or supplemented by other analyses (Stage E).

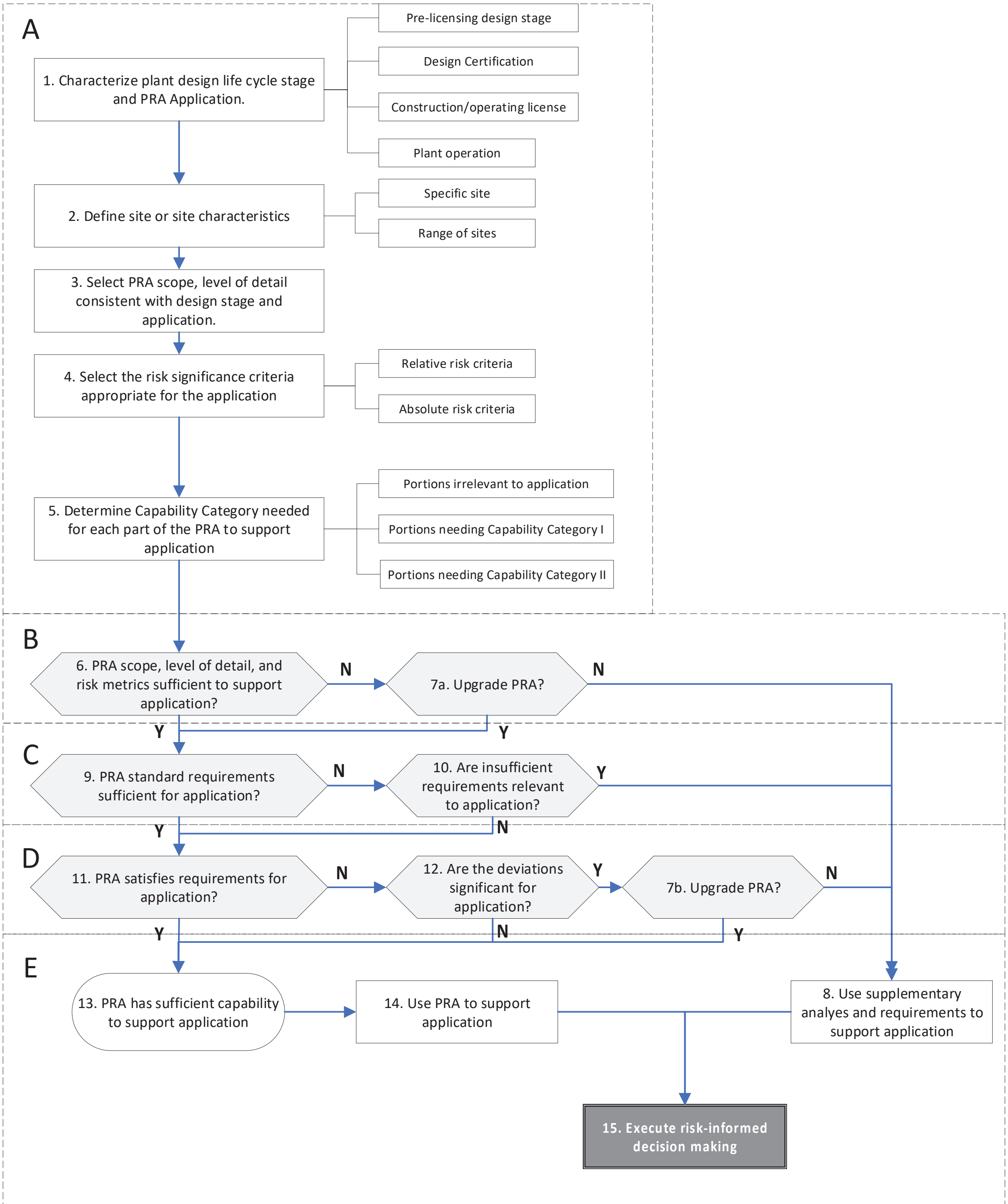
(c) Stage C. An evaluation is performed to determine whether the capability requirements for the SRs from this Standard for each relevant portion of the PRA are sufficient to support the application. If not, the SRs may be augmented with supplementary requirements as described in Stage E. For plants in a design stage, design requirements for the capability and reliability of SSCs may be augmented to address the deficiency.

(d) Stage D. Each relevant portion of the PRA is compared to the appropriate SRs in this Standard for the Capability Category needed to support the application as determined in Stage A. It is determined whether the relevant portions of the PRA have adequate capability, need upgrading to meet the appropriate set of SRs, or need supplementary analyses as described in Stage E. This stage also considers whether existing peer reviews are sufficient or whether additional peer reviews are required.

(e) Stage E. The relevant portions of the PRA, supplemented by additional analyses if necessary, are used to support the application. This activity is outside the scope of this Standard.

The scope of the activities in Figure 3-1 determines how to evaluate the role of the PRA in the application and how to determine which Capability Categories are needed for each portion of the PRA to support an application. The criteria for judging the quality of any supplementary analyses or requirements that are performed in lieu of upgrading the PRA to meet a desired Capability Category are application specific and are outside the scope of this Standard. Accordingly, to “meet this Standard” means that the portions of the PRA used in the application meet the High Level Requirements (HLRs) and SRs for a specified set of Capability Categories. The determination of how the PRA is used in the application and which technical requirements are applicable, and among those which Capability Categories are appropriate to the application, must be made on a case-by-case basis.

Fig. 3-1 PRA Application Flowchart



3.2 RANGE OF PRA APPLICATIONS CONSIDERED FOR THIS STANDARD

The range of applications that are envisioned for PRAs on advanced non-light water reactors (non-LWRs) is broader than has been the case for currently licensed reactors. In addition to the types of risk-informed applications that have been supported by PRAs on operating plants, the applications of PRAs for advanced non-LWRs may include the following:

(a) to evaluate alternative design options;

(b) to meet NRC requirements for design certification and licensing of new reactors;

(c) to support the safety classification of SSCs;

(d) to support the allocation of reliability and capability targets for SSCs;

(e) to help formulate the development of special treatment requirements;

(f) to support the evaluation of the adequacy of defense-in-depth.

Although the performance of each of these applications needs deterministic inputs, the availability of PRA technology provides the opportunity to perform risk-informed applications at an earlier stage of the plant life cycle than was possible for the existing reactors.

A number of approaches have been proposed for the use of PRA information for such applications as safety classification of SSCs and formulation of requirements for advanced non-LWR designs. One such approach was proposed for the modular high-temperature gas-cooled reactor (MHTGR) in the early 1990s that was subjected to a preliminary review by NRC [3-1]. Similar approaches have been proposed more recently for licensing the Pebble Bed Modular Reactor (PBMR) [3-2] and for use in developing a new American Nuclear Society (ANS) standard for designing MHTGRs [3-3]. NRC has developed a risk-informed and performance-based framework for future plant licensing that also involves the use of information from the PRA to support licensing applications [3-4]. The Licensing Modernization Project (LMP) [3-7] has produced a methodology that describes a risk-informed and performance-based approach for preparing license applications for advanced non-LWRs that builds on the above referenced approaches for the high-temperature gas-cooled reactors (HTGRs). While each of these approaches has its unique elements, all share the following characteristics of the approach outlined in NUREG-1860 [3-4]:

(a) All use a combination of deterministic and probabilistic elements and, hence, are classified as risk-informed approaches.

(b) Each approach uses the existing regulations and adjusts them to enable their application to non-LWR technology.

(c) Each approach uses information in the existing regulations (e.g., dose limits) to construct a frequency-versus-off-site radiological dose acceptance curve that is used to evaluate the acceptability of the combination of the frequencies and consequences of event sequences for the facility being analyzed. These acceptance curves are used to evaluate event sequence families having similar initiating events, plant responses in the satisfaction of safety functions, event progression, and end states. In addition to the evaluation of the individual risk contributors, the cumulative risks from the event sequences are compared against an accepted risk limit (e.g., Quantitative Health Objectives).

(d) Each approach uses information from the PRA to support the safety classification of SSCs and the derivation of special treatment requirements for SSCs.

(e) Each approach includes an evaluation of defense-in-depth including the application of deterministic defense-in-depth principles such as those in NUREG-0800, Chap. 19 [3-5], and NEI 18-04 [3-7] and consideration of the need for balance in the prevention and mitigation of event sequences.

3.3 STEP-BY-STEP APPROACH TO PRA APPLICATIONS

3.3.1 Stage A, Step 1. Characterize Plant Design-Life Cycle Stage and PRA Application (Box 1 in Figure 3-1)

PRA may be performed to support risk-informed decisions during any point in the plant design-life cycle. The purpose of this step is to define the design-life cycle stage and the associated PRA applications.

Examples of plant design-life cycle stages and associated applications include the following:

Life Cycle Stage	Example PRA Applications
Conceptual design	Evaluate candidate design options to fulfill reactor safety functions. Inform the selection of licensing basis events. Inform the safety classification of SSCs and selection of special treatment requirements. Set reliability and availability targets for SSCs. Evaluate the adequacy of defense-in-depth.
Design certification	Meet PRA requirements for design certification.
Combined operating license	Meet PRA requirements for license. Confirm that risk criteria are met.
Operating plant	Confirm that risk criteria are met. Evaluate risk-informed decisions.

3.3.2 Stage A, Step 2. Define Site Characteristics (Box 2 in Figure 3-1)

PRA may be performed to support risk-informed decisions that are site independent, for a bounding site, or for a specific site. These considerations impact the selection of PRA technical requirements from this Standard to support PRA applications.

Examples of site characteristics include the following:

Life Cycle Stage	PRA Site Characterization
Conceptual design	PRA performed for a plant design independent of site considerations.
Design certification	PRA performed for a plant design for a range of sites or a set of site characteristics that bounds a range of sites, i.e., a bounding site.
Combined operating license	PRA performed for a plant design at a specific site with well-defined site characteristics.

3.3.3 Stage A, Step 3. Select PRA Scope and Level of Detail Consistent With Design-Life Cycle Stage and Applications (Box 3 in Figure 3-1)

Define the application by the following:

(a) evaluating the plant design or operational change being assessed (Box 1 of Figure 3-1);

(b) characterizing the site characteristics (e.g., specific site or bounding site) (Box 2 of Figure 3-1);

(c) identifying the SSCs and plant activities affected by the change including the cause-effect relationship between the plant design or operational change and the PRA model (Box 1 of Figure 3-1);

(d) identifying the sources of radioactive material, plant operating states, hazard groups, PRA model scope, and PRA risk metrics that are needed to assess the change (Box 3 of Figure 3-1).

NUREG-0800 [3-5], EPRI TR-105396 [3-6], and NEI 18-04 [3-7] provide guidance for the above activities.

The scope and level of detail of a PRA is inherently limited by the level of detail in specifying the design, site, and operational characteristics of the plant and the intended PRA applications.

Examples of PRA scopes and levels of detail for different life cycle stages include the following:

Life Cycle Stage	Example PRA Applications and Associated Scope and Detail
Conceptual design	To develop a preliminary list of licensing-basis events, a simplified high-level PRA is performed based on a representative set of initiating events, assumed full-power initial conditions, functional event trees, simplified end states, and a qualitative evaluation of event sequence frequencies and consequences.
Design certification	To support a design certification application, a PRA is developed based on a detailed description of the front-line safety systems, comprehensive treatment of initiating events, a representative set of plant operating states, assumptions about capabilities support and balance-of-plant systems, simplified treatment of internal plant hazard events, seismic risk analysis and external hazards events analysis performed for a bounding site selected to cover a range of sites, mechanistic source terms, and off-site radiological doses.
Combined operating license	A full-scope PRA is performed for a plant design at a specific site with well-defined site characteristics, detailed design description of the entire plant, comprehensive treatment of initiating events and plant operating states, detailed analysis of internal plant hazards, seismic events, external plant hazards, mechanistic source terms, and off-site radiological consequences.

3.3.4 Stage A, Step 4. Select the Risk Significance Criteria Appropriate for the Application (Box 4 in Figure 3-1)

For applications involving the need for a calculation of baseline risk, relative risk significance criteria are selected. Such applications may require the PRA to determine the total integrated risk across all modeled sources of radioactive material, plant operating states, and hazard groups, whereas others may only require the evaluation of a specific combination of sources, plant operating states, and hazard groups. Applications that involve comparisons of the PRA results against fixed risk targets will need to use absolute risk significance criteria. Again, depending on the application, the PRA evaluation may require full or limited treatment of PRA model elements. See Section 1.9 for more discussion of risk significance criteria.

3.3.5 Stage A, Step 5. Determine Capability Category Needed for Each Portion of the PRA to Support Application (Box 5 in Figure 3-1)

The technical requirements in Section 4 set forth SRs for two Capability Categories whose attributes are described in Section 1.4. As explained in Section 1.4, the identification of the Capability Category is made in the context of the plant design-life cycle stage and the level of information available to describe the plant design and site at that design stage. For many of the SRs, the distinction between Capability Categories is based on the treatment of risk-significant contributors. Definitions in this Standard containing the words "risk significance" or "risk-significant" are generally written from the perspective of the total integrated risk, but "risk significance" may also be viewed in the context of a specific source of radioactive material, plant operating state, and hazard group. In addition, depending on the application, the definition of risk significance may be based on either absolute or relative risk metrics. It is important to recognize that for applications whose risk stems from more than one source of radioactive material, plant operating state, and hazard group, these definitions can be generalized to apply to the risk metrics that describe the total integrated risk or to specific parts of that risk. It is also recognized that for some applications, it is possible that the risk-significant contributors may encompass only a few event sequences or a small portion of a single hazard group.

"Risk significance" should also be treated differently for those SRs that refer to SRs in other hazard groups. For example, SR HR-G1 is incorporated by reference into the Human Reliability Analysis requirements of Internal Fire PRA. For an application in which the scope of the PRA would include a combination of internal events, internal plant hazards, and external events, a CC-II for this SR HR-G1 would require that the risk significance be defined in terms of the total integrated risk. In this case, the intent of CC-II would have been met.

For the application, determine the risk significance of each portion of the PRA for each hazard group needed to support the application (Box 5 of Figure 3-1). This risk significance determination dictates which Capability Category is needed for each SR for each portion of the PRA to support the application. To determine these capabilities, an evaluation shall be performed of the application to assess the role of the PRA in supporting that application including determining the relative importance of SRs to the application, identifying the portions of the hazard group PRA relevant to the application, and, for each relevant portion, determining the Capability Category for each SR needed to support the application. When performing this evaluation, the following application attributes shall be considered:

(a) role of the PRA in the application and extent of reliance of the decision on the PRA results;
(b) risk metrics to be used to support the application and associated decision criteria;
(c) relative or absolute risk significance of the risk contribution from each source of radioactive material, plant operating state, and hazard group to the decision;
(d) degree to which bounding or conservative methods for the PRA or in a given portion of the PRA would lead to inappropriately influencing the decisions made in the application, and approach(es) for accounting for this in the decision-making process;
(e) degree of accuracy and evaluation of uncertainties and sensitivities required of the PRA results;
(f) degree of confidence in the results that is required to support the decision;
(g) extent to which the decisions made in the application will impact the plant design basis.

The Capability Categories and the bases for their determination shall be documented.

Technical Specification (TS) Example[6]:

For an operating plant, a change in TSs is proposed that redefines the requirements for an operable service water (SW) system. This change removes the TS requirement for an allowed outage time (AOT) from one of the three pumps in each SW loop. In addition, the AOT for other selected combinations of inoperable components is increased. The changes in TSs and/or procedures that are involved need to be identified in detail.

To assess the impact of the proposed change in the TS, those SSCs, such as the SW system, affected by the proposed change need to be identified. The plant SW system has two redundant loops, each having two full-capacity SW pumps that use the ocean as the ultimate heat sink, and a third SW pump that uses a cooling tower (CT) and the atmosphere as the heat sink. The SW system is designed such that in the event of a reactor coolant system pressure boundary (RCB) breach concurrent with a loss of off-site power (LOOP), a single SW pump powered from its associated emergency diesel generator (EDG) will have sufficient capacity to meet the heat load. The existing TSs require two operable SW loops with each loop having three operable pumps. The proposed change redefines an operable SW loop as having one operable SW pump and one operable CT SW pump, removes the AOT requirements from two SW pumps, lengthens the AOT requirement for SW pumps in the same loop to bring it into line with the AOT for single SW train unavailability, and increases the standby CT SW pump AOT based on its lower risk importance.

The proposed change in the AOT impacts the frequency of certain release categories by increasing the likelihood that an SW pump would be unavailable due to planned or

[6] The examples in this Standard are focused primarily on internal events. Additional examples will be added in a future revision.

unplanned maintenance. This change is evaluated by considering the impact on system unavailability and on the frequency of sequences making a risk-significant contribution and involving unavailability of a single train of SW.

Continuation of Example:

The proposed change is a risk-informed application to justify a change to an operating license. It is determined that the change in each release event sequence family frequency due to changes in SW maintenance unavailability associated with the proposed TS change is not significant. Therefore, the portions of the PRA that are impacted by changes in SW pump unavailability due to maintenance are determined to require CC-II, whereas the remaining portions of the PRA needed to determine the overall plant risk are determined to require only CC-I. Hence, the SRs for plant operating states, initiating events, event sequences, data parameters, system models, human actions, and quantification process for those sequences and cutsets impacted by the AOT changes are in PRA CC-II, and the SRs for the remaining portions of the PRA needed to evaluate each release category frequency (RCF) are in CC-I.

Non-LWR Design Certification Example:

Consider the example of a PRA that is performed for an advanced non-LWR in support of a design certification application. Because the PRA is done during a design stage, the level of detail of the PRA models is limited to that supported by the available design details. The PRA scope that is required addresses all internal and external hazard groups, plant operating states, and each source of radioactive material. Because the PRA is performed for a set of site characteristics selected to cover a range of sites, the treatment of external hazards must use external hazard curves that bound the range of sites associated with the design certification. Some of the SRs cannot be met due to lack of an as-built and as-operated plant, and the associated limitations of the PRA would then need to be documented to meet other SRs in this Standard. Because of the nature of the application, SRs associated with model completeness and uncertainties are determined to require CC-II. However, because of the design-stage aspect of the PRA, not all SRs will require CC-II. For example, based on the expected frequency distribution for RCB breaches, the SRs for Systems Analysis for RCB breaches below a certain size may require CC-II, while breaches larger than that can be analyzed using a bounding calculation, consistent with CC-I.

3.3.6 Stage B. PRA Scope and Risk Metrics Sufficient to Support Application (Boxes 6 and 7a in Figure 3-1)

Determine whether the PRA provides the results needed to assess the plant or operational change (Box 6 of Figure 3-1). If some aspects of the PRA are insufficient to assess the change, then upgrade them in accordance with the SRs in the technical requirements in Section 4 for its corresponding Capability Category (Box 7a of Figure 3-1) or generate supplementary analyses (see Section 3.3.7).

If it is determined that the PRA is sufficient, the bases for this determination shall be documented. Any upgrade of the PRA shall be done and documented in accordance with the requirements in Section 5. If peer review is required or needs to be updated, it is done with the general peer review requirements in Section 6 and the technical element specific peer review requirements in Section 4.

Example:

The proposed change in the SW AOT has been determined to affect the SW unavailability during at-power operating conditions. For the plant in question, the SW provides cooling to the residual heat removal (RHR) pumps, the diesel generators, the feedwater pumps, the component cooling water (CCW) system, and the radioactive waste system. Therefore, for internal events, the scope of the internal initiating events at-power analysis element of the PRA must include the following:

(a) RCB breach initiators, since the change in SW unavailability will affect the plant capabilities to mitigate RCB breach events;

(b) LOOP initiators, since the SW change will affect the availability of the plant diesel generators;

(c) loss-of-feedwater initiators, since the feedwater pumps are SW cooled.

Although the SW cools the CCW system, there is enough thermal inertia in the CCW system to allow it to function for several hours after the loss of SW, thereby enabling the plant to be placed in a safe stable state; a loss of CCW initiator would not be needed for this application. Also, since the radioactive waste system does not make a risk-significant contribution to any event sequence family frequency even when the SW system is assumed to be unavailable, it need not be considered in evaluating the PRA application. Any impact would be considered in Box 15 of Figure 3-1, as needed. It is determined that the changes in maintenance unavailability are too small to consider impacts on the reliability of the SW pumps that could impact a wider range of sequences, including loss of SW initiating events and sequences with SW pump failures. These impacts are combined in the plant model to calculate the change in each event sequence family frequency. A determination is made that there are no risk-significant contributions to any event sequence family frequencies associated with SW pump unavailability initiated by external hazards, and hence, only the portions of the PRA associated with internal events need to be considered in this application.

Next, it is necessary to determine if the SSCs or plant activities affected by the PRA application are modeled in the PRA (Box 6 of Figure 3-1). If the affected SSCs or plant activities are not modeled, then either upgrade the PRA to include the SSCs in accordance with the SRs in the technical requirements in Section 4 for their corresponding Capability Category (Box 7a of Figure 3-1) or generate supplementary analyses (see Section 3.3.7).

If it is determined that each portion of the PRA is sufficient, the bases for this determination shall be documented. Any upgrade of the PRA shall be done and documented in accordance with Section 5. If peer review is required or needs to be updated, it is done with the general peer review requirements in Section 6 and the technical element specific peer review requirements in Section 4.

Example:
Continuing with the previous TS example, the SSCs and plant activities related to the systems impacted by the proposed change in the SW, and that contribute to the change in event sequence family frequencies (i.e., RCB breach mitigation system, diesel generators, feedwater, and CCW), need to be modeled in the PRA. For example, if the loss-of-feedwater initiator is modeled as one global initiator, then either the PRA needs to be upgraded to include the relationship between SW and feedwater, or the effect of SW on feedwater must be resolved by using supplementary analyses outside of this Standard. (See Box 8 of Figure 3-1.)

As part of this step of the applications process, the portions of a PRA that are needed for an application shall have been reviewed pursuant to the requirements for peer review in Section 6.

3.3.7 Stage C. Requirements in PRA Standard Sufficient for Application (Boxes 9 and 10 in Figure 3-1)

The purpose of this step is to determine whether the scope of coverage and level of detail of the SRs stated in the technical requirements in Section 4 for the corresponding Capability Categories determined in Section 3.3.5 are sufficient to assess the application under consideration (Box 9 of Figure 3-1).

If it is determined that this Standard lacks specific requirements, their relevance to the application shall be assessed (Box 10 of Figure 3-1). If the absent requirements are not relevant, the requirements of this Standard are sufficient for the application. The bases for determining the sufficiency of this Standard shall be documented. If the absent requirements are relevant, supplementary requirements may be used (Box 8 of Figure 3-1).

3.3.8 Stage D. PRA Satisfies Requirements for Application (Boxes 11, 12, and 7b in Figure 3-1)

The purpose of this step is to determine whether each portion of the PRA satisfies the SRs at the appropriate Capability Category needed to support the application (Box 11 of Figure 3-1). The results of the peer review may be used. If the PRA meets the SRs necessary for the application, the PRA is acceptable for the application being considered (Box 13 of Figure 3-1). The bases for this determination shall be documented.

If the PRA does not satisfy an SR for the appropriate Capability Category, then determine whether the reason it is not being satisfied is relevant or significant (Box 12 of Figure 3-1). Acceptable requirements for demonstrating the relevance or significance include either of the following:

(a) The reason for not meeting the SR at the appropriate Capability Category is not relevant if it is not applicable or does not affect quantification relative to the impact of the proposed application [for example, if an SR related to the treatment of human reliability has not been met because some of the human error probabilities for human failure events (HFEs) that are significant in the base case have not been evaluated using a detailed Human Reliability Analysis method, but those particular HFEs play no role in the results needed for the application, then the failure to meet CC-II is not relevant to the decision].

(b) The difference is not significant if the modeled event sequences accounting for at least 90% of each event sequence family frequency associated with all the sources of radioactive material, plant operating states, and hazard groups being evaluated, as applicable, are not adversely affected by appropriate sensitivity studies or bounding evaluations. These studies or evaluations should measure the aggregate impact of the exceptions to the requirements in the technical requirements in Section 4 relevant to the application. The relevant hazard groups may be evaluated separately or in a combined fashion, as needed to determine the significance of the difference for the application.

This determination will depend on the particular application being considered and may involve determinations made by an expert panel.

If the difference is neither relevant nor significant, then the PRA is acceptable for the application. If the difference is relevant or significant, then either upgrade the PRA to address the corresponding SRs stated in the technical requirements in Section 4 (Box 7b of Figure 3-1) or generate supplementary analyses (see Section 3.3.7). Any upgrade of the PRA shall be done and documented in accordance with the requirements in Section 5.

3.3.9 Stage E. Use of PRA in Decision-Making (Boxes 8, 13, 14, and 15 in Figure 3-1)

If the scope of either the PRA or this Standard is insufficient, supplementary analyses or requirements may be used (Box 8 of Figure 3-1). These supplementary analyses will depend on the particular application being considered—but may involve deterministic methods such as bounding or screening analyses—and determinations made by an expert panel. They shall be documented.

Example of Supplementary Analysis:
A change in testing frequency is desired for motor-operated valves (MOVs) judged to be of low safety significance by using a risk-informed ranking method. Not all MOVs or MOV failure modes of interest within the program are represented in the PRA. Specifically, valves providing an isolation function between the reactor vessel and low-pressure piping may only be represented in the frequency of an initiating event associated with RCB breaches bypassing a radionuclide transport barrier. The inadequate PRA model representation can be supplemented by categorizing the group of high-pressure/low-pressure-interface MOVs in an appropriate event sequence family frequency model.

The categorization is based on PRA insights that indicate failure of MOVs to isolate reactor vessel have the potential to lead to a barrier bypass condition. This example illustrates a process of addressing SSC model adequacy by using general risk information to support the placement of MOVs into the appropriate risk category.

Supplementary requirements shall be drawn from other recognized codes or standards whose scopes complement that of this Standard and that are applicable to the application but may be generated by an expert panel if no such recognized code or standard can be identified.

Example of Supplementary Requirements:

A risk ranking/categorization for a plant's in-service inspection (ISI) program is being pursued. The current PRA model meets the requirements set forth in this Standard. However, this Standard does not provide requirements for modeling piping or pipe segments adequate to support a detailed quantitative ranking. This Standard can be supplemented with an expert panel to determine the safety significance of pipe segments. Considerations of deterministic and other traditional engineering analyses, defense-in-depth philosophy, or maintenance of safety margins could be used to categorize pipe segments. Use of published industry or NRC guidance documents on risk-informed ISI could also be used to supplement this Standard. The PRA model could also be used to supplement this Standard by estimating the impact of each pipe segment's failure on risk without modifying the PRA's logic. This estimate could be accomplished by identifying an initiating event, basic event, or group of events, already modeled in the PRA, whose failures capture the effects of the pipe segment failure.

Second Example of Supplementary Requirements:

It is desired to rank the snubbers in a plant according to their risk significance for developing a graded approach to snubber testing. With the exception of snubbers on large primary system components, snubbers have been shown to have a small impact on frequency of radionuclide release; therefore, this Standard does not require their failure to be addressed in determining RCFs. However, testing programs are required to demonstrate the capability of snubbers to perform their dynamic support function. As shown in ASME Code Case OMN-10 [3-8], evaluation of failure mechanisms may show that the safety significance of snubbers can be approximated by the safety significance of the components that they support for the events in which the snubbers are safety significant, and this supplementary criterion could be used to rank the safety importance of the snubbers.

Third Example of Supplementary Requirements:

It is desired to replace certain MOVs that are currently considered safety-grade with commercial-grade equipment when new valves are procured. The internal events PRA shows that these valves have a minor role in important event sequences and that the only important failure mode is failure to open on demand. The failure rate of the commercial-grade valves for this mode is known through reliable data to be identical to the failure rate for safety-grade valves. However, the question arises about whether the commercial-grade valves will perform as well as safety-grade valves during and after a large earthquake. To address this question, supplementary requirements, found in an appropriate reference (e.g., Nuclear Engineering and Design [3-9]) may be used. By using this reference, the seismic capacity of the commercial-grade valves can be evaluated and can be compared to that of the safety-grade valves that they would replace.

If it has been determined that the PRA has sufficient capability, its results can be used to support the application (Box 14 of Figure 3-1). If not, the results of supplementary analyses, some of which may respond to supplementary requirements, can also be used to support the application (Box 8 of Figure 3-1). Such supplementary analyses/requirements are outside the scope of this Standard.

The risk contributors and associated uncertainties should be characterized for all risk metrics, plant operating states, release categories, event sequence families, and hazard groups selected as relevant to the application. Once all relevant contributions to risk have been characterized, the risk input is provided for the RIPB decision (Box 15 of Figure 3-1). The relevant sources of radioactive material, plant operating states, and hazard groups may be characterized separately or in a combined fashion, as needed to support the application.

For risk-informed applications, the terms "relevance" or "risk significance" can be evaluated from different perspectives. "Relevance" is related to the applicability of a certain part of the PRA such as source of radioactive material, release category, hazard group, etc. "Risk significance" of event sequences, contributors, cutsets, etc., can be measured either by their contribution to a specific part of the PRA model such as source of radioactive material, plant operating state, or hazard group (e.g., fires within the plant), by their contribution to the overall plant risk, or by comparison to a fixed risk target. Hence, depending on the application, either relative or absolute risk significance criteria may be appropriate. The characterization of risk contributors from part of the PRA model (e.g., hazard group) can occur in the development, approval, and implementation phases of risk-informed applications. When performing activities intended to meet this Standard at the level of an HLR or an SR, use of the term "risk significance" requires an assessment and characterization of the contribution of risk contributors to the overall risk, risk within the scope of the hazard groups relevant to the application, or risk compared to fixed risk targets.

When performing a PRA using this Standard, addressing “risk significance” requires an assessment and characterization of the relative contribution of risk contributors within a given source of radioactive material, plant operating state, and hazard group as well as for the overall plant risk. For example, an SR that identifies an action to be performed for “significant” fire zones is assessed within the context of the other fire risk contributors for the full-power plant operating state only (i.e., within the part of the PRA for internal fires initiated during full-power operation). However, when performing a risk-informed application, it is often more appropriate to evaluate “risk significance” across all relevant sources of radioactive material, plant operating states, and hazard groups and to use absolute rather than relative risk significance criteria. When the risk-informed application is implemented, it is necessary to determine whether it would alter baseline assumptions or plant conditions such that “risk significance” within a specific portion of the PRA model is now altered or more uncertain. The evaluation of “significance” at this level may or may not require further analysis within a specific portion of the PRA model.

In meeting the requirements (HLRs and SRs) of this Standard, those SRs associated with assessing or identifying levels of risk significance are first performed for the PRA models that are used to quantify average annual estimates of risk from all modeled sources of radioactive material, plant operating states, and hazard groups. With regard to the applications process of this Standard, the assessment, or identification of significance (Box 12 of Figure 3-1), is to be evaluated first across all risk contributors within the context of the change(s) being proposed by the risk-informed application and then within each relevant radioactive material source, plant operating state, and hazard group to determine whether additional analysis is necessary.

3.4 REFERENCES

The following is a list of publications referenced in this Standard.

[3-1] NUREG-1338, “Draft Preapplication Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor,” U.S. Nuclear Regulatory Commission, March 1989

[3-2] “NRC Staff’s Preliminary Findings Regarding Exelon Generation’s (Exelon’s) Proposed Licensing Approach for the Pebble Bed Modular Reactor (PBMR),” U.S. Nuclear Regulatory Commission, March 26, 2002

[3-3] ANSI/ANS-53.1-2011 (R2016), “Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants,” American Nuclear Society, 2011

[3-4] NUREG-1860, “Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing,” U.S. Nuclear Regulatory Commission, December 2007

[3-5] NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants,” Chap. 19, “Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decision-Making: General Guidance,” U.S. Nuclear Regulatory Commission, 1998

[3-6] EPRI TR-105396, D. TRUE et al., “PSA Applications Guide,” Electric Power Research Institute, August 1995

[3-7] NEI 18-04, “Risk-Informed Performance-Based Technology Inclusive Guidance for Advanced Reactor Licensing Basis Development,” Nuclear Energy Institute, Revision 1, August 2019

[3-8] ASME OM Code for Operation and Maintenance of Nuclear Power Plants, Code Case OMN-10, “Requirements for Safety Significance Categorization of Snubbers Using Risk Insights and Testing Strategies for Inservice Testing of LWR Power Plants,” American Society of Mechanical Engineers, 2020

[3-9] R. P. KENNEDY and M. K. RAVINDRA, “Seismic Fragilities for Nuclear Power Plant Risk Studies,” Nucl. Eng. Des., 79, 1, 47, 1984

SECTION 4 - RISK ASSESSMENT TECHNICAL REQUIREMENTS CONTENTS

Nonmandatory Appendices

(The text presented in blue font in this standard comprise hyperlinks to enable efficient access to referenced sections and elements, requirements, notes, references, etc.)

SECTION 4
RISK ASSESSMENT TECHNICAL REQUIREMENTS

4.1 PURPOSE

The purpose of this Section is to provide technical requirements for a PRA for an advanced non-light water reactor (non-LWR) nuclear power plant (NPP). This Section also includes general requirements for use of expert judgment in performance of the PRA.

4.2 USE OF EXPERT JUDGMENT

This paragraph provides requirements for the use of expert judgment outside of the PRA analysis team to resolve a specific technical issue.

Guidance from NUREG/CR-6372 [4-1] and NUREG-1563 [4-2] may be used to meet the requirements in this Section. Other approaches, or a mix of these, may also be used.

Examples:

Use of expert judgment to resolve difficult issues includes Pacific Gas and Electric's Diablo Canyon seismic study [4-3] and the Yucca Mountain project's study of volcanic hazards [4-4]. These reports provide useful insights into both the strengths and the potential pitfalls of using experts. A review of expert-aggregation methods, the different types of consensus, and issues with resolving disagreements among experts can be found in Appendix J of NUREG/CR-6372 [4-1].

4.2.1 Objective of Using Expert Judgment

The PRA analysis team shall explicitly and clearly define the objective of the information that is being sought through the use of outside expert judgment and shall explain this objective and the intended use of the information to the expert(s).

4.2.2 Identification of the Technical Issue

The PRA analysis team shall explicitly and clearly define the specific technical issue to be addressed by the expert(s).

4.2.3 Determination of the Need for Outside Expert Judgment

The PRA analysis team may elect to resolve a technical issue using their own expert judgment or the judgment of others within their organization.

The PRA analysis team shall use outside experts when the needed expertise on the given technical issue is not available within the analysis team or within the team's organization. The PRA analysis team should use outside experts, even when such expertise is available inside, if there is a need to obtain broader perspectives, for any of the following or related reasons:

(a) complex experimental data exist that the analysts know have been interpreted differently by different outside experts;

(b) more than one conceptual model exists for interpreting the technical issue, and judgment is needed as to the applicability of the different models;

(c) judgments are required to assess whether bounding assumptions or calculations are appropriately conservative;

(d) uncertainties are large and risk-significant, and judgments of outside technical experts are useful in illuminating the specific issue.

4.2.4 Identification of Expert Judgment Process

The PRA analysis team shall determine the following:

(a) the degree of importance and the level of complexity of the issue;

(b) whether the process will use a single entity (individual, team, company, etc.) that will act as an evaluator and integrator and will be responsible for developing the community distribution or will use a panel of expert evaluators and a facilitator/integrator.

The facilitator/integrator shall be responsible for aggregating the judgments and community distributions of the panel of experts so as to develop the composite distribution of the informed technical community.

4.2.5 Identification and Selection of Evaluator Experts

The PRA analysis team shall identify one or more experts capable of evaluating the relative credibility of multiple alternative hypotheses to explain the available information. These experts shall evaluate all potential hypotheses and bases of inputs from the literature, and from proponents and resource experts, and shall provide the following:

(a) their own input;

(b) their representation of the community distribution.

4.2.6 Identification and Selection of Technical Issue Experts

If needed, the PRA analysis team shall also identify other technical issue experts, such as

(a) experts who advocate particular hypotheses or technical positions (e.g., an individual who evaluates data and develops a particular hypothesis to explain the data);

(b) technical experts with knowledge of a particular technical area of relevance to the issue.

4.2.7 Responsibility for the Expert Judgment

The PRA analysis team shall assign responsibility for the resulting judgments, either to an integrator or shared with the experts. Each individual expert shall accept responsibility for his or her individual judgments and interpretations.

4.2.8 Expert Judgment References

The following is a list of publications referenced in this Standard.

[4-1] NUREG/CR-6372, R. J. BUDNITZ, G. APOSTOLAKIS, D. M. BOORE, L. S. CLUFF, K. J. COPPERSMITH, C. A. CORNELL, and P. A. MORRIS, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and the Use of Experts," U.S. Nuclear Regulatory Commission, 1997

[4-2] NUREG-1563, J. P. KOTRA, M. P. LEE, N. A. EISENBERG, and A. R. DeWISPELARE, "Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program," U.S. Nuclear Regulatory Commission, 1996

[4-3] "Final Report of the Diablo Canyon Long Term Seismic Program," Pacific Gas and Electric Company; available from the U.S. Nuclear Regulatory Commission, Dockets 50-275 and 50-323, 1988

[4-4] BA000-1717-2200-00082, "Probabilistic Volcanic Hazards Analysis for Yucca Mountain, Nevada," U.S. Department of Energy Yucca Mountain Project, 1996

[4-5] ASME/ANS RA-S-1.4-2013, "Probabilistic Risk Assessment Standard for Advanced non-LWR Nuclear Power Plants," Trial Use for Pilot Applications Standard, American Society of Mechanical Engineers and American Nuclear Society, 2013

[4-6] ASME/ANS RA-Sb-2013 (R2019), "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers and American Nuclear Society, 2013

[4-7] NEI 18-04, "Risk-Informed Performance-Based Technology Inclusive Guidance for Advanced Reactor Licensing Basis Development," Nuclear Energy Institute, Revision 1, August 2019

4.3 TECHNICAL REQUIREMENTS

4.3.1 Plant Operating State Analysis (POS)

This Section presents the technical requirements associated with Plant Operating State Analysis.

4.3.1.1 Objectives and Technical Requirements for Plant Operating State Analysis

The objectives of the Plant Operating State Analysis ensure that

(a) each plant operating state so identified is to be defined in terms of all important conditions that may affect the delineation and evaluation of event sequences modeled in the PRA;

(b) plant operating states that are grouped together are shown to be represented by the characteristics of the remaining group;

(c) the frequencies, decay heat levels, and plant configurations for each plant operating state are well characterized and may be assumed to be constant or time-varying within the Plant Operating State Analysis, depending on how finely the plant operating states are defined;

(d) document the Plant Operating State Analysis to provide traceability of the work.

Table 4.3.1.1-1 High Level Requirements for Plant Operating State Analysis

Designator	Requirement
HLR-POS-A	The Plant Operating State Analysis shall use a structured, systematic process to identify and define plant operating states to be considered in the PRA, consistent with the specific reactor design and scope of the PRA.
HLR-POS-B	The Plant Operating State Analysis shall justify all screening and grouping of plant operating states or plant evolutions to facilitate an efficient estimation of event sequence frequencies and event sequence family frequencies.
HLR-POS-C	The Plant Operating State Analysis shall determine the plant operating state frequencies and durations along with the representative decay heat levels and other plant parameters relevant to the frequency or consequences of event sequences and event sequence families.
HLR-POS-D	The documentation of the Plant Operating State Analysis shall provide traceability of the work.

Table 4.3.1.1-2 Supporting Requirements for HLR-POS-A

The Plant Operating State Analysis shall use a structured, systematic process to identify and define plant operating states to be considered in the PRA, consistent with the specific reactor design and scope of the PRA. (HLR-POS-A)

Index No. POS-A	Capability Category I	Capability Category II
POS-A1	IDENTIFY a representative set of plant evolutions to be analyzed. INCLUDE, at a minimum, plant evolutions from at-power operations. See Note POS-N-1, POS-N-2, POS-N-3, POS-N-4	IDENTIFY a representative set of plant evolutions to be analyzed, including refueling outages, other controlled shutdowns, and forced outages. See Note POS-N-3
POS-A2	For each identified plant evolution, REVIEW available plant- or design-specific documentation and records for the following: *(a)* operating modes or operational conditions; *(b)* reactor coolant boundary (RCB) configurations, such as vented or not vented; whether temporary reactor coolant system (RCS) penetrations are installed and their differential pressure capability changes in configuration of vessel internals; and decay heat removal (DHR) mechanisms; *(c)* range of RCS parameters (e.g., power level or decay heat level, average reactor coolant temperatures, pressures, and coolant inventories, and other parameters needed to determine success criteria, mechanistic source terms, and radiological consequences); *(d)* available instrumentation for key parameters to be monitored for the plant operating states; *(e)* activities that may lead to changes in the above parameters used to define the plant operating states; *(f)* status of radionuclide transport barriers; *(g)* activities changing the capabilities of structures, systems, and components (SSCs) to support each of the reactor-specific safety functions that must be satisfied during each plant operating state; *(h)* operational assumptions on full power, shutdown, refueling, and startup conditions. See Note POS-N-5, POS-N-6, POS-N-7, POS-N-8	
POS-A3	For each plant evolution, DEFINE the characteristics of a set of exclusive plant operating states that cover the entire evolution in terms of unique combinations of the following non-exhaustive list: *(a)* sources of radioactive material within the scope of the PRA, including non-core sources, unless there is a technical justification as documented per POS-D1 for excluding it; *(b)* operating modes or operational conditions; *(c)* RCB configurations; *(d)* the range of RCS parameters and the selected representative parameter value chosen for each plant operating state (e.g., power level or decay heat level including typical plant operating state entry times after plant trip, average reactor coolant temperatures, pressures, and coolant inventories, and other parameters needed to determine success criteria, mechanistic source terms, and radiological consequences); *(e)* available parameter instrumentation for defining the plant operating states; *(f)* activities that may lead to changes in the above parameters used to define the plant operating states; *(g)* status of radionuclide transport barriers. ENSURE the set of plant operating states is sufficient to support the selection of initiating events, the justification of success criteria, plant operating state frequency and duration parameters, the evaluations of human failure events (HFEs), the accounting for planned equipment outages, the definition of event sequences, and the quantification of sequence frequency, and to provide a finite number of sets of plant conditions for peer reviews. See Note POS-N-5, POS-N-6, POS-N-9, POS-N-10, POS-N-11	
POS-A4	For operating plants, ENSURE the level of detail in the delineation of plant operating states is consistent with the as-built and as-operated plant sufficient to identify potential risk-significant contributors. See Note POS-N-12	

Table 4.3.1.1-2 Supporting Requirements for HLR-POS-A (Cont'd)

The Plant Operating State Analysis shall use a structured, systematic process to identify and define plant operating states to be considered in the PRA, consistent with the specific reactor design and scope of the PRA. (HLR-POS-A)

Index No. POS-A	Capability Category I	Capability Category II
POS-A5	For PRAs performed during the pre-operational stage, ENSURE the level of detail in delineating the plant operating states is consistent with the level of detail of the design information available to support, and referenced by the PRA sufficient to identify potential risk-significant contributors. See Note POS-N-13	
POS-A6	For operating plants, REVIEW known plans for future plant evolutions (e.g., the next refueling outage) to ensure the selections made in Requirement POS-A3 remain valid and appropriate. As a minimum, consider the following: *(a)* plant operating states involving higher risk that were not previously encountered but are expected in future plant evolutions; *(b)* earlier entry into a plant operating state, resulting in substantially higher decay heat, or later entry into a plant operating state, resulting in substantially lower decay heat; *(c)* durations of plant operating states (see Requirement POS-C1). See Note POS-N-12	
POS-A7	For operating plants, INTERVIEW appropriate plant personnel to determine whether potential plant operating state analysis of past or future evolutions have been overlooked. See Note POS-N-12, POS-N-14, POS-N-15, POS-N-16	
POS-A8	For PRAs performed during the pre-operational stage, INTERVIEW knowledgeable engineering personnel to confirm that the selection of plant operating states correctly represents the as-designed and as-intended-to-operated plant. See Note POS-N-13, POS-N-14, POS-N-15	
POS-A9	ENSURE that *(a)* the plant conditions defined for each plant operating state of the modeled plant evolutions allow the analysis to meet the requirements of the remaining PRA elements, as applicable; *(b)* the plant operating state intervals and the selected parameter conditions are sufficient to identify risk-significant contributors.	
POS-A10	If the scope of the PRA includes hazard groups other than internal events, REVIEW the plant conditions defined for each plant operating state to ENSURE that they remain sufficient for those hazard groups to do the following: *(a)* support the selection of initiating events, the justification of success criteria, plant operating state frequency and duration parameters, the evaluations of HFEs, the accounting for planned equipment outages, and the quantification of event sequence frequencies; *(b)* provide a finite number of sets of plant conditions for peer reviews. See Note POS-N-17	
POS-A11	For each plant operating state defined in POS-A3, IDENTIFY the SSCs and their desired operational characteristics needed to support safe operation in that plant operating state. See Note POS-N-18	
POS-A12	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with plant operating state definition in a manner that supports the applicable requirements of HLR-ESQ-E.	
POS-A13	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built and as-operated details that influence plant operating state definitions. See Note POS-N-13, POS-N-19	

Table 4.3.1.1-3 Supporting Requirements for HLR-POS-B

The Plant Operating State Analysis shall justify all screening and grouping of plant operating states or plant evolutions to facilitate an efficient estimation of event sequence frequencies and event sequence family frequencies. (HLR-POS-B)

<table>
<tr><th>Index No. POS-B</th><th>Capability Category I</th><th>Capability Category II</th></tr>
<tr><td>POS-B1</td><td>GROUP plant evolutions into a set of representative evolutions.
ENSURE that
(a) the evolutions within a group can be considered similar in terms of the set of plant operating states that they contain; and
(b) the evolutions are bounded by the worst case impact within the group.</td><td>GROUP plant evolutions into a set of representative evolutions.
ENSURE that
(a) the evolutions can be considered similar in terms of the set of plant operating states that they contain;
(b) the evolutions are bounded by the worst case impact within the group;
(c) the grouping does not impact risk-significant event sequences; and
(d) the impact from each evolution is comparable to those of the remaining evolutions in the group.</td></tr>
<tr><td>POS-B2</td><td colspan="2">RETAIN plant operating states unless they can be screened out by satisfying the requirement of SCR-1, SCR-2, or SCR-3 in Table 1.10-1.
JUSTIFY the use of alternate frequency or consequence screening criteria.</td></tr>
<tr><td>POS-B3</td><td colspan="2">GROUP plant operating states appearing in the same plant evolution type and considered similar in terms of plant response, success criteria, frequency, and the effect on the performance of operator and operability and performance of relevant mitigating systems.
ENSURE the grouped plant operating states do not mask risk-significant contributors or risk insights.</td></tr>
<tr><td>POS-B4</td><td colspan="2">ENSURE that plant operating states with different plant response impacts (e.g., those with different success criteria or radionuclide transport barrier configurations) or those that could have more severe radionuclide release potential remain separated.</td></tr>
<tr><td>POS-B5</td><td>GROUP plant operating states that involve initiating events that are “demand based” with initiators that are time based.</td><td>SEPARATE plant operating states that are used for those brief time periods involving activities (test-, maintenance-, and evolution-related) that lead to initiating events that are “demand based” from those that are time based.
If necessary, DELINEATE such plant operating states to avoid averaging the short duration of the demand over an entire plant operating state duration or, if needed, to ensure that the representative plant conditions defined for the plant operating state apply at the time of the “demand-based” initiating events.</td></tr>
<tr><td>POS-B6</td><td colspan="2">If plant operating states are combined into groups, ENSURE that the most severe or constraining characteristics (with respect to the frequency and consequences of each event sequence family, including consideration of the type and frequency of initiating events) of any plant operating state within the group are chosen for the combined group.</td></tr>
<tr><td>POS-B7</td><td colspan="2">IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with plant operating state screening and grouping in a manner that supports the applicable requirements of HLR-ESQ-E.</td></tr>
<tr><td>POS-B8</td><td colspan="2">For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built and as-operated details that influence plant operating state screening and grouping.
See Note POS-N-13</td></tr>
</table>

Table 4.3.1.1-4 Supporting Requirements for HLR-POS-C

The Plant Operating State Analysis shall determine the plant operating state frequencies and durations along with the representative decay heat levels and other plant parameters relevant to the frequency or consequences of event sequences and event sequence families. (HLR-POS-C)

Index No. POS-C	Capability Category I	Capability Category II
POS-C1	Within the selected plant evolutions, CALCULATE the mean duration and the mean time after shutdown for each plant operating state based on a review of applicable plant- or design-specific record. See Note POS-N-7, POS-N-8, POS-N-20	
POS-C2	For PRAs performed during the pre-operational stage, PROVIDE the basis for the assumed mean duration and time in the plant operating cycle of each modeled plant operating state. See Note POS-N-13	
POS-C3	SUM the durations for each group of plant operating states to obtain the durations of the groups. The entry frequencies of the grouped plant operating states are the same (see Requirement POS-B1), though the frequencies differ for each type.	
POS-C4	For plant evolutions involving low power and shutdown (LPSD), CALCULATE the decay heat level associated with each plant operating state for use in defining and applying success criteria and the timing for operator actions.	
POS-C5	REVIEW known future plans or upcoming evolution schedules to ensure the quantification of assumed decay heat levels and plant operating state durations remain valid. See Note POS-N-12	

Table 4.3.1.1-5 Supporting Requirements for HLR-POS-D

The documentation of the Plant Operating State Analysis shall provide traceability of the work. (HLR-POS-D)

Index No. POS-D	Capability Category I	Capability Category II
POS-D1	DOCUMENT the process used in the Plant Operating State Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of Supporting Requirements (SRs) are satisfied: *(a)* selection and definitions of the plant evolutions; *(b)* the process and criteria used to identify plant operating states; *(c)* the process and criteria used to group plant operating states; *(d)* the definition of each plant operating state group; *(e)* the defining characteristics of each plant operating state; *(f)* the mean durations, mean times since shutdown, and mean frequencies of plant operating states; *(g)* the decay heat associated with each plant operating state of each plant evolution; *(h)* specific interfaces with other PRA tasks for traceability and to facilitate configuration control when interfacing tasks are updated. See Note POS-N-21	
POS-D2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirements POS-A12 and POS-B7) associated with the Plant Operating State Analysis.	
POS-D3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Plant Operating State Analysis. See POS-A13 and POS-B8 See Note POS-N-13	

4.3.1.2 Peer Review Requirements for Plant Operating State Analysis

4.3.1.2.1 Purpose

This Section provides requirements for peer review of the Plant Operating State Analysis element of the PRA.

4.3.1.2.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the areas of plant operating state definition, Initiating Event Analysis, and Event Sequence Analysis. The team members assigned to review Event Sequence Analysis shall overlap those assigned to review event sequence and Initiating Event Analysis to ensure consistency between the modeling for these elements. The team members assigned to review the Plant Operating State Analysis shall have experience specific to these areas and the capability of recognizing plant-specific features of the analyses.

4.3.1.2.3 Review of Plant Operating State Analysis to Confirm the Methodology

A review shall be performed on the entire Plant Operating State Analysis. The Plant Operating State Analysis review typically includes the following:

(a) definition of the plant operating states for the plant PRA;

(b) review adequacy of the plant operating state definition for the stage of design intended PRA applications;

(c) plant evolutions selected for the PRA;

(d) modeling and quantification of the fraction of time spent in each plant operating state;

(e) identification of important dependencies for consideration in the Initiating Event Analysis and Event Sequence Analysis.

4.3.1.3 References for Plant Operating States Analysis

The following is a list of publications referenced in this Standard.

[POS-1] NUREG/CR-6143, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1," U.S. Nuclear Regulatory Commission, June 1994

[POS-2] NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1," U.S. Nuclear Regulatory Commission, October 1995

[POS-3] EPRI TR-1003113, "An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989–2000): Outage Risk Assessment and Management (ORAM) Technology," Electric Power Research Institute, 2001

[POS-4] IAEA-SSG-3, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants Specific Safety Guide: IAEA Safety Standards Series No. SSG-3," International Atomic Energy Agency, 2010

[POS-5] NSAC-84, D. C. BLEY, J. W. STETKAR, and L. A. BOWEN, "Zion Nuclear Plant Residual Heat Removal PRA," Electric Power Research Institute Nuclear Safety Analysis Center, July 1985

NONMANDATORY APPENDIX POS: NOTES AND EXPLANATORY MATERIAL FOR PLANT OPERATING STATE ANALYSIS

POS.1 NOTES ASSOCIATED WITH PLANT OPERATING STATE ANALYSIS

Table POS-1 Notes Supporting Plant Operating State Analysis Requirements

Number	Notes
POS-N-1	An example of plant evolution from at-power operations is taking down train(s) of operation for maintenance while operating at power. See POS-A1
POS-N-2	Early pre-operational stage PRAs are typically limited to at-power PRAs only. See POS-A1
POS-N-3	Examples of plant evolutions include power changes (e.g., load-following, transitions to low power or shutdowns) and transitions to maintenance configurations, refueling outages, and forced outages. See POS-A1
POS-N-4	The plant operating states and plant evolutions to be analyzed depend on the PRA scope, pre-operational stage, and application. Depending on the application, the evolution to be addressed may range from at-power only to all plant operating states outage types. See POS-A1
POS-N-5	Examples of activities that may lead to changes in the parameters used to define the plant operating states include drain down, filling and venting, dilution, fuel movement, and cooldown. See POS-A2, POS-A3
POS-N-6	Examples of radionuclide transport barriers include reactor building, containment, or confinement. Examples of status of radionuclide transport barriers include deinerted, intact, and open. See POS-A2, POS-A3
POS-N-7	For operating plants, examples of available records for review are recent outage plans and records, maintenance plans and records, operations data, trip history, and control room logbooks. Examples of documentation include Technical Specifications, and normal shutdown, refueling, and startup procedures. See POS-A2, POS-C1
POS-N-8	For PRAs performed during the pre-operational stage, the records and documentation will exist in preliminary form in design documents supplemented by assumptions. See POS-A2, POS-C1
POS-N-9	Examples of sources of radioactive material besides that in the reactor core that may be included in the scope of the PRA include radioactive material circulating or plated out within the RCB, spent fuel in the spent fuel storage system, fuel/salt processing systems, radioactive waste systems, and other process systems with radioactive material. See POS-A3
POS-N-10	The combination of all plant operating states covers all of the modeled plant evolutions. See POS-A3
POS-N-11	The more detailed set of plant conditions selected may, for example, be expanded to include all those needed to compute a time-dependent risk profile of plant configurations for a plant- or design-specific evolution (e.g., system alignments and equipment out of service). When documenting the Plant Operating State Analysis, it is not necessary to explicitly list each of the plant operating states and the specific values of the selected plant conditions in an exhaustive table of all plant operating states . For example, the user may choose to describe the selection of initiating events applicable to each of the plant operating states in terms of the subset of plant conditions that apply (e.g., RCS pressure and level conditions). Such explanations should cover all of the plant operating state plant conditions listed in Requirement POS-A3. See POS-A3

Table POS-1 Notes Supporting Plant Operating State Analysis Requirements (Cont'd)

Number	Notes
POS-N-12	This SR is likely not applicable to PRAs performed during the pre-operational stage. See POS-A4, POS-A6, POS-A7, POS-C5
POS-N-13	This SR is not applicable to operating plants. See POS-A5, POS-A8, POS-A13, POS-B8, POS-C2, POS-D3
POS-N-14	Examples of appropriate detailed reviews include tabletop reviews and computerized walkdowns. See POS-A7, POS-A8
POS-N-15	Examples of appropriate plant personnel interviewed include representatives from operations, maintenance, engineering, safety analysis, and outage planning. See POS-A7, POS-A8
POS-N-16	Information from interviews conducted at similar plants may also be used but are not a substitute for plant-specific interviews. See POS-A7
POS-N-17	To ensure the plant conditions defined for each plant operating state are sufficient for hazard groups other than internal events, consider changing plant conditions that may impair or change the effectiveness of hazard barriers, affect propagation pathways, or modify fragilities of SSCs. See POS-A10
POS-N-18	Examples of SSCs and their desired operational characteristics are valve X fully closed, pump Y in standby, and pump Z running. See POS-A11
POS-N-19	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See POS-A13
POS-N-20	This requirement can be addressed in conjunction with Requirements POS-A1, POS-A2, and POS-A4. See POS-C1
POS-N-21	The set of plant operating states is the organizing structure for the definition of event sequences modeled in the PRA. See POS-D1

4.3.2 Initiating Event Analysis (IE)

This Section presents the technical requirements associated with Initiating Event Analysis.

4.3.2.1 Objectives and Technical Requirements for Initiating Events Analysis

The objectives of the Initiating Event Analysis ensure that

(a) there is a reasonably complete identification of initiating events;
(b) grouping the initiating events is conducted so that events in the same group have similar mitigation requirements;
(c) frequencies of the initiating events are quantified; and
(d) the Initiating Event Analysis is documented to provide traceability of the work.

Table 4.3.2.1-1 High Level Requirements for Initiating Event Analysis

Designator	Requirement
HLR-IE-A	The Initiating Event Analysis shall reasonably identify all initiating events for all modeled plant operating states and sources of radioactive material consistent with the PRA scope and plant pre-operational stage.
HLR-IE-B	The Initiating Event Analysis shall group the initiating events so that events in the same group have similar mitigation requirements (i.e., the requirements for all events in the group are either equally or less restrictive than the limiting mitigation requirements for the group) to facilitate an efficient but realistic estimation of the frequency of each modeled event sequence and event sequence family.
HLR-IE-C	The Initiating Event Analysis shall quantify the annual frequency of each initiating event or initiating event group based on the plant conditions for each source of radioactive material and plant operating state within the scope of the PRA.
HLR-IE-D	The documentation of the Initiating Event Analysis shall provide traceability of the work.

Table 4.3.2.1-2 Supporting Requirements for HLR-IE-A

The Initiating Event Analysis shall reasonably identify all initiating events for all modeled plant operating states and sources of radioactive material consistent with the PRA scope and plant pre-operational stage. (HLR-IE-A)

Index No. IE-A	Capability Category I	Capability Category II
IE-A1	IDENTIFY those initiating events that challenge normal plant operation, when the plant is at-power, or the ability to sustain safe shutdown or low-power conditions, when not at-power, and that require successful mitigation to prevent a release of radioactive material. USE a structured, systematic process for identifying initiating events that accounts for plant- or design-specific features. See Note IE-N-1, IE-N-2	
IE-A2	For each source of radioactive material selected within the scope of the PRA, IDENTIFY mechanisms by which this material could be mobilized to escape barriers identified in Requirement MS-B2 including mechanisms associated with all the hazard groups selected within the scope of the PRA. ENSURE that identified mechanisms are included in the selection of initiating events for the PRA. USE a structured, systematic process for identifying mechanisms that accounts for plant- or design-specific features. See Note IE-N-1, IE-N-2	
IE-A3	For operating plants, ENSURE the level of detail in the delineation of initiating events is consistent with the as-built and as-operated plant sufficient to identify potential risk-significant contributors. See Note IE-N-3, IE-N-4	
IE-A4	For PRAs performed during the pre-operational stage, ENSURE the level of detail in delineating the initiating events is consistent with the level of detail of the available design information referenced by the PRA. See Note IE-N-5	

Table 4.3.2.1-2 Supporting Requirements for HLR-IE-A (Cont'd)

The Initiating Event Analysis shall reasonably identify all initiating events for all modeled plant operating states and sources of radioactive material consistent with the PRA scope and plant pre-operational stage. (HLR-IE-A)

Index No. IE-A	Capability Category I	Capability Category II
IE-A5	INCLUDE in the spectrum of initiating event challenges at least the following general categories: *(a)* transient INCLUDE among the transient category both equipment- and human-induced events that disrupt the plant and leave the reactor coolant system boundary (RCB) intact. DELINEATE transient initiators in a manner that resolves the unique challenges to the reactor-specific safety functions; *(b)* RCB breach INCLUDE in the RCB breach category both equipment- and human-induced events that disrupt the plant by causing a breach in the reactor coolant system (RCS) with a resulting loss of coolant inventory or pressure. DELINEATE the RCB breach initiators, using a defined rationale for the differentiation; *(c)* interfacing systems RCB breaches INCLUDE postulated events in systems interfacing with the RCS that could fail or be operated in such a manner as to result in an uncontrolled loss of core coolant or release of radioactive material bypassing a radionuclide transport barrier as applicable; *(d)* special initiators; *(e)* any of the above caused by an internal plant hazard; *(f)* any of the above caused by an external hazard; *(g)* other categories of initiating events caused by at-initiator human failure events. See Note IE-N-1, IE-N-6, IE-N-7, IE-N-8, IE-N-9, IE-N-10, IE-N-11, IE-N-12	
IE-A6	When identifying initiating events caused by internal or external hazards, INCLUDE initiating events caused by a combination of hazards (e.g., seismically induced fires, flooding caused by fire sprinkler actuation) included in the scope of the PRA.	
IE-A7	For operating plants, REVIEW the plant-specific initiating-event experience to ensure that the list of challenges addresses plant experience for all modeled plant operating states. See also Requirement IE-A11 See Note IE-N-4	
IE-A8	REVIEW generic analyses of similar plants to assess whether the list of challenges included in the model addresses industry experience for all modeled plant operating states. If no similar plants can be reviewed, AUGMENT the review for all major modeled systems in the PRA to include generic analyses from other facilities' comparable systems (e.g., systems at non-LWRs, LWRs, nonnuclear facilities, fuel cycle facilities) to assess whether the list of challenges included in the model addresses relevant industry experience for all modeled plant operating states to review every major modeled system in the PRA. See also Requirement IE-A11 See Note IE-N-13, IE-N-14, IE-N-15	REVIEW generic analyses and operating experience of similar plants to assess whether the list of challenges included in the model addresses industry experience for all modeled plant operating states. If no similar plants can be reviewed, AUGMENT the review for all major modeled systems in the PRA to include generic analyses and operating experience from other facilities' comparable systems (e.g., systems at non-LWRs, LWRs, nonnuclear facilities, fuel cycle facilities) to assess whether the list of challenges included in the model addresses for relevant industry experience for all modeled plant operating states to review every major modeled system in the PRA. See also Requirement IE-A11 See Note IE-N-13, IE-N-14, IE-N-15

Table 4.3.2.1-2 Supporting Requirements for HLR-IE-A (Cont'd)

The Initiating Event Analysis shall reasonably identify all initiating events for all modeled plant operating states and sources of radioactive material consistent with the PRA scope and plant pre-operational stage. (HLR-IE-A)

Index No. IE-A	Capability Category I	Capability Category II
IE-A9	PERFORM a systematic evaluation of each system down to the subsystem or train level and including support systems in each modeled plant operating state, to assess the possibility of an initiating event occurring due to a failure of the system or train. PERFORM a qualitative review of system impacts to identify potential system initiating events.	PERFORM a systematic evaluation of each system down to the subsystem or train level and including support systems in each modeled plant operating state, to assess the possibility of an initiating event occurring due to a failure of the system or train. USE a structured approach [such as a system-by-system review of initiating event potential, Hazards and Operability Study (HAZOPS), or failure modes and effects analysis (FMEA), or other systematic process] to assess and document the possibility of an initiating event resulting from individual systems or train failures.
IE-A10	When performing the systematic evaluation required in Requirement IE-A9, INCLUDE in each modeled plant operating state initiating events resulting from multiple failures if the equipment failures result from a common cause.	When performing the systematic evaluation required in Requirement IE-A9, INCLUDE in each modeled plant operating state initiating events resulting from multiple failures, including equipment failures resulting from random or common causes or equipment unavailabilities involving routine system alignments for preventive or corrective maintenance or testing configurations.
IE-A11	In the identification of the initiating events, INCLUDE *(a)* events that have occurred at this plant and similar plants or systems, as applicable, during any plant operating state conditions that could also occur during the plant operating state identified for each modeled initiating event; *(b)* events that have occurred at this plant and similar plants or systems, as applicable, resulting in an unplanned controlled shutdown that includes a scram prior to reaching low-power conditions, unless it is determined that an event is not applicable to the plant operating state identified for each modeled initiating event. See Note IE-N-14	
IE-A12	INTERVIEW at least one resource knowledgeable in plant design or operation to determine if potential initiating events have been overlooked.	EVALUATE the identified initiating events for the plant conditions via investigation(s) depending on the plant design-life cycle stage of the PRA for identifying additional initiating events. See Note IE-N-16
IE-A13	For operating plants, REVIEW the operating experience for initiating event precursors, for the purposes of identifying additional initiating events, and to identify initiating events caused by human failures that impact later operator mitigation actions for all applicable plant operating states. See Note IE-N-4, IE-N-17, IE-N-18	
IE-A14	For PRAs performed during the pre-operational stage, if similar plants exist, REVIEW operating experience with similar plants for initiating event precursors, for the purposes of identifying additional initiating events, and to identify initiating events caused by at-initiator human failures that impact later operator mitigation actions for all applicable plant operating states. See Note IE-N-5, IE-N-17, IE-N-18, IE-N-19	

Table 4.3.2.1-2 Supporting Requirements for HLR-IE-A (Cont'd)

The Initiating Event Analysis shall reasonably identify all initiating events for all modeled plant operating states and sources of radioactive material consistent with the PRA scope and plant pre-operational stage. (HLR-IE-A)

Index No. IE-A	Capability Category I	Capability Category II
IE-A15	In searching for initiating events, INCLUDE each system and supporting system alignment, such as temporary alignments during maintenance, that could either influence the likelihood that failures cause an initiating event or could increase the severity of the effect on plant safety functions that would result from such an event. It is acceptable to include such cases by subsuming them within other cases to be analyzed to facilitate the analysis if documented and justified.	
IE-A16	For PRAs performed on plants with two or more reactors or modular reactor plants, INCLUDE initiators that impact two or more reactors or sources of radioactive material (e.g., multi-reactor loss-of-off-site-power events or total loss of service water) that may impact the model. IDENTIFY the number and specific combination of reactors and sources of radioactive material that are impacted by each selected initiator. See Note IE-N-20, IE-N-21	
IE-A17	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with initiating event identification in a manner that supports the applicable requirements of HLR-ESQ-E.	
IE-A18	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the initiating event identification analysis. See Note IE-N-5, IE-N-22	

Table 4.3.2.1-3 Supporting Requirements for HLR-IE-B

The Initiating Event Analysis shall group the initiating events so that events in the same group have similar mitigation requirements (i.e., the requirements for all events in the group are either equally or less restrictive than the limiting mitigation requirements for the group) to facilitate an efficient but realistic estimation of the frequency of each modeled event sequence and event sequence family. (HLR-IE-B)

Index No. IE-B	Capability Category I	Capability Category II
IE-B1	GROUP initiating events to facilitate definition of event sequences in the Event Sequence Analysis element and to facilitate quantification in the Event Sequence Quantification element. JUSTIFY that grouping initiating events does not affect the determination of risk-significant event sequences in the different plant operating states or sources of radioactive material within the scope of the PRA. See Note IE-N-23	
IE-B2	USE a structured, systematic process for grouping initiating events. See Note IE-N-24	
IE-B3	For PRAs performed during the pre-operational stage, GROUP the initiating events at a level of detail that is consistent with the design information available, sufficient to identify potential risk-significant contributors. See Note IE-N-5	

Table 4.3.2.1-3 Supporting Requirements for HLR-IE-B (Cont'd)

The Initiating Event Analysis shall group the initiating events so that events in the same group have similar mitigation requirements (i.e., the requirements for all events in the group are either equally or less restrictive than the limiting mitigation requirements for the group) to facilitate an efficient but realistic estimation of the frequency of each modeled event sequence and event sequence family. (HLR-IE-B)

Index No. IE-B	Capability Category I	Capability Category II
IE-B4	GROUP initiating events only when *(a)* events can be considered similar in terms of plant response, success criteria, timing, and the effect on performance of operators and the operability of relevant mitigating systems; or *(b)* events can be bounded by the worst-case impacts within the group.	GROUP initiating events only when *(a)* events can be considered similar in terms of plant response, success criteria, timing, and the effect on performance of operators and the operability of relevant mitigating systems; or *(b)* events can be bounded by the worst-case impacts within the group and the grouping does not impact risk-significant event sequences.
IE-B5	For multi-reactor plants, IDENTIFY initiating events in different initiating event groups that impact specific combinations of multiple reactors and sources of radioactive material in shared facilities consistent with the scope of the PRA. See Note IE-N-21	
IE-B6	For PRAs performed during the pre-operational stage, IDENTIFY assumptions due to the lack of as-built and as-operated details that impact initiating event groupings. See Note IE-N-5, IE-N-22	

Table 4.3.2.1-4 Supporting Requirements for HLR-IE-C

The Initiating Event Analysis shall quantify the annual frequency of each initiating event or initiating event group based on the plant conditions for each source of radioactive material and plant operating state within the scope of the PRA. (HLR-IE-C)

Index No. IE-C	Capability Category I	Capability Category II
IE-C1	For operating plants, CALCULATE the initiating event frequency by addressing applicable generic and plant- or design-specific data that is representative of current design and performance unless it is justified that there are adequate plant-specific data to characterize the parameter value and its uncertainty. See Note IE-N-4, IE-N-25, IE-N-26, IE-N-27, IE-N-28	
IE-C2	For PRAs performed during the pre-operational stage and thus with no operating experience, CALCULATE the initiating event frequency by addressing applicable generic and design-specific data that is representative of current design and proposed performance and available data from similar plants. See Note IE-N-5, IE-N-25, IE-N-26, IE-N-27, IE-N-28	
IE-C3	For operating plants, when using generic or plant-specific data, USE data representative of current design and performance to quantify the initiating event frequencies. See Note IE-N-4	
IE-C4	INCLUDE recovery actions (those implied in Requirement IE-C9 and those implied and discussed in Requirements IE-C11, IE-C12, IE-C14, and IE-C15) as appropriate. JUSTIFY each recovery action (as evidenced through procedures or training). See Note IE-N-4	
IE-C5	For PRAs performed during the pre-operational stage, INCLUDE recovery actions as appropriate. JUSTIFY each recovery action (e.g., as evidenced by the plants operating philosophy) in a manner that is consistent with the level of detail of the design and operations information. See Note IE-N-5	

Table 4.3.2.1-4 Supporting Requirements for HLR-IE-C (Cont'd)

The Initiating Event Analysis shall quantify the annual frequency of each initiating event or initiating event group based on the plant conditions for each source of radioactive material and plant operating state within the scope of the PRA. (HLR-IE-C)

Index No. IE-C	Capability Category I	Capability Category II
IE-C6	For operating plants, when combining evidence from generic and plant-specific data, USE a Bayesian update process or equivalent statistical process. JUSTIFY the selection of any informative prior distribution used based on industry experience. See Note IE-N-4, IE-N-29	
IE-C7	For PRAs performed during the pre-operational stage, IDENTIFY, REVIEW, and USE (as appropriate when sufficient applicable nuclear power plant data are not available) relevant, available, and applicable experience from other facilities (potentially including nonnuclear facilities, fuel cycle facilities, and nonpower reactors) in the quantification of initiating event frequencies. See Note IE-N-5	
IE-C8	CALCULATE initiating event frequencies on a plant-calendar-year basis, where a plant may be comprised of two or more reactors or sources of radioactive material. INCLUDE the fraction of time the plant is in the applicable plant operating state in the initiating event frequency calculation. See Note IE-N-30	
IE-C9	USE screening criteria at least as stringent as the following characteristics to eliminate initiating events or groups from further evaluation. If other screening criteria are used, ensure that they meet the criteria in Table 1.10-1 and that the bases are justified (as demonstrated per Requirement ESQ-D8): *(a)* the event does not involve either a failure or bypass of a radionuclide transport barrier; *(b)* either: *(1)* the event has the same impact on the plant as another event that has a much higher frequency per the requirements of SCR-1 or SCR-2 in Table 1.10-1, or *(2)* the event does not require the plant to go to shutdown conditions until sufficient time has expired during which the initiating-event conditions, with a high degree of certainty (based on supporting calculations), are detected and corrected (either administratively or automatically) such that a complicated shutdown does not occur per the requirements of SCR-3 in Table 1.10-1.	
IE-C10	ENSURE data represents plant design and operational performance (if applicable). See Note IE-N-31	ENSURE data represents plant design and operational performance (if applicable). JUSTIFY data that were excluded as neither current nor applicable (e.g., provide evidence via design or operational change that the data are no longer applicable). See Note IE-N-31
IE-C11	If fault tree modeling is used for initiating events, USE the applicable Systems Analysis requirements for fault tree modeling found in Systems Analysis (HLR-SY-A). INCLUDE in the modeling of initiating event fault trees the contribution of human failure events (HFEs) during test, maintenance, and other plant evolution activities leading to the initiating event or PROVIDE the basis for exclusion. See Note IE-N-32	

Table 4.3.2.1-4 Supporting Requirements for HLR-IE-C (Cont'd)

The Initiating Event Analysis shall quantify the annual frequency of each initiating event or initiating event group based on the plant conditions for each source of radioactive material and plant operating state within the scope of the PRA. (HLR-IE-C)

Index No. IE-C	Capability Category I	Capability Category II
IE-C12	If fault tree modeling is used for initiating events, MODIFY, as necessary, the fault tree computational methods that are used so that the top event quantification produces a failure frequency rather than a top event probability as normally computed. QUANTIFY the initiating event frequency (as opposed to the probability of an initiating event over a specific time frame), which is the usual fault tree quantification model described in Systems Analysis (SRs in HLR-SY-A). USE the applicable requirements in Data Analysis (SRs in HLR-DA-C and HLR-DA-D) for the data used in the fault tree quantification and the SRs of Human Reliability Analysis (SRs in HLR-HR-G) for human reliability input to the Initiating Event Analysis. See Note IE-N-33	
IE-C13	For those initiating events involving operator directions from the control room [identified in IE-A5(g)], ESTIMATE the contribution of operators in the control room to the initiating event frequency. For an initiating event group that has a mixture of hardware and operator contributions: ESTIMATE the split fraction or percentage of each initiating event that has historically been caused, or is estimated to be caused, by operators in the control room, and ESTIMATE the split fraction or percentage of each initiating event that does not involve operators in the control room. See Note IE-N-34	
IE-C14	If fault tree modeling or human reliability analysis is used for initiating events, INCLUDE within the initiating event fault tree models all relevant combinations of events involving the annual frequency of one component failure combined with the unavailability (or failure during the repair time of the first component) of other components.	
IE-C15	If fault tree modeling or human reliability analysis is used for initiating events, USE plant- or design-specific information, as applicable, in the assessment and quantification of recovery actions where available, consistent with the applicable requirements in Human Reliability Analysis.	
IE-C16	Where plant- or design-specific data are used, COMPARE result with generic data sources, and EXPLAIN differences in the Initiating Event Analysis to provide a reasonableness check of the results.	
IE-C17	For rare initiating events, USE industry generic data, and INCLUDE plant- or design-specific functions. If no applicable industry events have occurred, expert judgment may be used; if used, AUGMENT with applicable generic data sources and SATISFY the requirements of Section 4.2, Use of Expert Judgment.	For rare initiating events, USE industry generic data, and AUGMENT with a plant- or design-specific fault tree or other evaluation that addresses unique plant- or design-specific features. If no applicable industry events have occurred, expert judgment may be used; if used, AUGMENT with applicable generic data sources and SATISFY the requirements of Section 4.2, Use of Expert Judgment. ADDRESS in the quantification the plant- or design-specific features that could influence initiating events and recovery probabilities.

Table 4.3.2.1-4 Supporting Requirements for HLR-IE-C (Cont'd)

The Initiating Event Analysis shall quantify the annual frequency of each initiating event or initiating event group based on the plant conditions for each source of radioactive material and plant operating state within the scope of the PRA. (HLR-IE-C)

Index No. IE-C	Capability Category I	Capability Category II
IE-C18	In the analysis of RCB breaches that bypass a radionuclide transport barrier due to faults in interfacing systems, INCLUDE the following features of plant and procedures that influence the initiating event frequency: *(a)* configuration of potential pathways including numbers and types of values and their relevant failure modes, the existence, size, positioning of relief valves, and behavior of other components (e.g., pump seals, heat exchangers, etc.); *(b)* provision of protective interlocks; *(c)* relevant surveillance test procedures; *(d)* the capability of interfacing system piping; *(e)* isolation capabilities given high flow/differential pressure conditions that might exist following breach of the interfacing system.	
IE-C19	CALCULATE a point estimate for the initiating event frequencies. CHARACTERIZE the uncertainty for those initiating event frequencies associated with risk-significant event sequences. See Note IE-N-35	CALCULATE a mean value for the frequencies of the risk-significant initiating events. PROVIDE the probabilistic representation of the uncertainty of the parameter estimates of risk-significant initiating events. Acceptable methods include Bayesian updating or expert judgment. If using expert judgment, SATISFY the requirements of Section 4.2, Use of Expert Judgment. For the non-risk-significant initiating events, ENSURE the requirement for Capability Category I (CC-I) is met.

Table 4.3.2.1-5 Supporting Requirements for HLR-IE-D

The documentation of the Initiating Event Analysis shall provide traceability of the work. (HLR-IE-D)

Index No. IE-D	Capability Category I	Capability Category II
IE-D1	DOCUMENT the process used in the Initiating Event Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the functional categories considered and the specific initiating events included in each; *(b)* the systematic search for plant-unique and plant- or design-specific support system initiators; *(c)* the approach to identify initiating events specific to each modeled source of radioactive material and plant operating state; *(d)* the systematic search for RCB failures for each size and location with unique challenges to the reactor-specific safety functions; *(e)* the approach for assessing completeness and consistency of initiating events with previous experience (e.g. plant- or design-specific experience, industry experience, other comparable PRAs, etc.); *(f)* the basis for screening out initiating events; *(g)* the basis for grouping and subsuming initiating events including the basis for characterizing the impact of the initiating event on single or multiple reactor plants or non-core radionuclide sources; *(h)* justification of the dismissal of any observed initiating events; *(i)* the derivation of the initiating event frequencies and the recoveries used; *(j)* the approach to quantification of each initiating event frequency including the basis for the fraction of time in each applicable plant operating state; *(k)* the justification for exclusion of any data; *(l)* the justification for applying events or data collected for other types of reactors or facilities.	
IE-D2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirements IE-A17) associated with the Initiating Event Analysis.	
IE-D3	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Initiating Event Analysis. See IE-A18 and IE-B6 See Note IE-N-5	

4.3.2.2 Peer Review Requirements for Initiating Events Analysis

4.3.2.2.1 Purpose

This section provides requirements for peer review of the Initiating Event Analysis element of the PRA.

4.3.2.2.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the area of initiating event analysis including experience in performing systematic search for initiating events in a new reactor technology and methods for such purpose including plant level FMEA, master logic diagrams, and process hazards analysis. The team members assigned to review Initiating Event Analysis shall overlap the team assigned to review Event Sequence Analysis to ensure consistency between the modeling for both elements. The team members assigned to review the Event Sequence Analysis shall have experience specific to these areas and the capability of recognizing plant-specific features of the analyses.

4.3.2.2.3 Review of Initiating Event Analysis to Confirm the Methodology

The entire Initiating Event Analysis shall be reviewed.

4.3.2.3 References for Initiating Events Analysis

The following is a list of publications referenced in this Standard.

[IE-1] NUREG/CR-6823, "Handbook of Parameter Estimation for Probabilistic Risk Assessment," Sandia National Laboratories/U.S. Nuclear Regulatory Commission, September 2003

[IE-2] NUREG/CR-5750, "Rates of Initiating Events at U.S. Nuclear Power Plants: 1987–1995," Idaho National Engineering and Environmental Laboratory, February 1999

[IE-3] EPRI TR-1003113, "An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989–2000): Outage Risk Assessment and Management (ORAM) Technology," Electric Power Research Institute, 2001

[IE-4] International Atomic Energy Agency, Technical Approach to Probabilistic Safety Assessment for Multiple Reactor Units, Safety Series Report No. 96, 2019

[IE-5] Safety Reports Series No. 96, "Technical Approach to Probabilistic Safety Assessment for Multiple Reactor Units," International Atomic Energy Agency, 2019

NONMANDATORY APPENDIX IE: NOTES AND EXPLANATORY MATERIAL FOR INITIATING EVENTS ANALYSIS

IE.1 NOTES ASSOCIATED WITH INITIATING EVENTS ANALYSIS

Table IE-1 Notes Supporting Initiating Event Analysis Requirements

Number	Notes
IE-N-1	Depending on the scope of the PRA and its application, the SRs for this High Level Requirement (HLR) may apply to different plant operating states and different sources of radioactive material in the plant. It is necessary to define what is meant by “normal” plant operation for each plant operating state. Once normal plant operation for a plant operating state is defined, events are identified that challenge normal operation. It is recognized that these initiating events are conditional on the plant operating state of each selected plant evolution within the scope of the PRA. Many initiating events for at-power conditions will also apply to plant operating states with the RCS at operating pressure. Special emphasis is placed on review of plant evolutions and maintenance activities (including plant realignments in preparation for maintenance) during shutdown plant operating states to identify initiating events unique to these operating conditions. See IE-A1, IE-A2, IE-A5
IE-N-2	For example, such a systematic approach to identifying initiating events may employ master logic diagrams, heat balance fault trees, process hazards analysis (PHA), or FMEA. Human-initiated events may be postulated by analyzing the operating procedures and practices applicable for each plant operating state consistent with the scope of the PRA. Existing lists of known initiators applicable to the specific reactor type and design are also commonly employed as a starting point. See IE-A1, IE-A2
IE-N-3	For operating plants, the basis for judging the adequacy of the level of detail, fidelity, and realism of the PRA is the available set of characteristics of the as-built and as-operated plant. See IE-A3
IE-N-4	This SR is likely not applicable to PRAs performed during the pre-operational stage. See IE-A3, IE-A7, IE-A13, IE-C1, IE-C3, IE-C4, IE-C6
IE-N-5	This SR is not applicable to operating plants. See IE-A4, IE-A14, IE-A18, IE-B3, IE-B6, IE-C2, IE-C5, IE-C7, IE-D3
IE-N-6	Examples in the transient category include transients with the addition of positive core reactivity (e.g., core rod withdrawal) and without the addition of positive core reactivity (e.g., spurious reactor trip). See IE-A5
IE-N-7	Examples of external hazards include seismic events and aircraft crashes. See IE-A5
IE-N-8	Examples of initiating events caused by at-initiator human failure events may include the following: *(a)* those initiating events that have been historically caused by an operator in the control room (if historical data is available); *(b)* those initiating events that involve the control room operators providing directions (e.g., RCS drain down for liquid-cooled reactors); *(c)* those initiating events that do not involve receiving directions from operators in the control room, such as component failures. See IE-A5
IE-N-9	Examples of internal plant hazards include internal flood, internal fire, high-energy line break, or human error. See IE-A5

Table IE-1 Notes Supporting Initiating Event Analysis Requirements (Cont'd)

Number	Notes
IE-N-10	Examples of RCB breach types include the following: *(a)* RCB breaches of different sizes and locations. Examples: pipe breaks of different sizes and locations; *(b)* excessive RCB breaches: include RCB breaches in sizes or locations that cannot be adequately mitigated within the design basis by any combination of inherent reactor characteristics or engineered systems. Example: reactor pressure vessel rupture; *(c)* RCB heat exchanger failures; *(d)* unisolated RCB breaches outside or bypassing a radionuclide transport barrier; *(e)* RCB breaches unique to specific plant operating states; and *(f)* RCB breaches specific to an internal or external event hazard (e.g., internal fire or seismic event). See IE-A5
IE-N-11	Examples of special initiators include support systems failures, instrument line breaks, and events specific to a source of radioactive material. See IE-A5
IE-N-12	Initiating events are defined at the level of the plant disturbance (e.g., transient or RCB breach), but each initiating event may be caused by an internal event, another internal hazard (e.g., fire or flood), or external hazard. Requirements for identifying initiating events due to specific hazards are available for internal floods, internal fires, seismic events, high winds, external flooding, and other hazards. For each hazard, there is the potential for new initiating events to be identified for the PRA model and, hence, the references in this requirement to initiating events due to specific hazards. See IE-A5
IE-N-13	Generic analysis in this context is a systematic analysis that can inform the development of initiating events (e.g., FMEAs). See IE-A8
IE-N-14	Similar plants can be taken broadly for new reactors without any directly comparable plants by reviewing operating experience at the system level. For example, molten salt reactors may review pebble bed high temperature gas reactors for online refueling plant operating state insights and related challenges. The search can be widened to include similar systems at nonnuclear facilities, fuel cycle facilities, and nonpower reactors if needed to ensure insights from this review cover the majority of in-scope plant systems. See IE-A8, IE-A11
IE-N-15	The extended review is not intended to be endless. Once a review has been performed for every major system modeled in the PRA at a facility or facilities that utilize a comparable system, the extended review may stop. See IE-A8
IE-N-16	Examples of investigations include, but are not limited to, actives such as: tabletop review(s), physical/computerized walkdown(s) of the plant, and/or interview(s) with knowledgeable personnel. These investigation(s) may need to be supported by a review of plant documents, especially in the context of external hazard evaluations. This SR does not imply that each of these activities needs to be performed to meet the SR. Further, if appropriate, one investigation can meet the intent of multiple investigation SRs if the investigation is scoped and structured appropriately. See IE-A12
IE-N-17	For example, plant- or design-specific experience with intake structure clogging might indicate that loss of intake structures should be identified as a potential initiating event. See IE-A13, IE-A14
IE-N-18	For the purpose of this requirement, an initiating event precursor is an event or condition that is assessed to increase the probability of an imminent initiating event by over an order of magnitude. See IE-A13, IE-A14
IE-N-19	For PRAs performed during the pre-operational stage, it is assumed that operating experience does not exist. If similar plants also do not exist, this requirement is not applicable. See IE-A14

Table IE-1 Notes Supporting Initiating Event Analysis Requirements (Cont'd)

Number	Notes
IE-N-20	Guidance on the modeling of single and multi-reactor initiating events in multi-reactor PRAs is found in [IE-4] and [IE-5]. See IE-A16
IE-N-21	This SR is not applicable to PRAs performed for a single reactor plant. See IE-A16, IE-B5
IE-N-22	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See IE-A18, IE-B6
IE-N-23	During certain plant evolutions such as those involving low power and shutdown, a variety of system configurations may be entered. An initiating event grouping valid for one plant operating state may not be appropriate for another. Identifying the "bounding" or "worst case" also requires a careful review of plant operating practices and may require an iterative process as insights for preliminary modeling become available. See IE-B1
IE-N-24	For example, such a systematic approach for grouping initiating events may employ master logic diagrams, heat balance fault trees, PHA methods such as HAZOPs, or FMEA. See IE-B2
IE-N-25	A useful reference for generic initiating event frequencies is NUREG/CR-5750 [IE-2]. See IE-C1, IE-C2
IE-N-26	A useful reference for initiating event frequencies during shutdown is EPRI TR-1003113 [IE-3]. See IE-C1, IE-C2
IE-N-27	A useful reference for parameter estimation methodology is NUREG/CR-6823 [IE-1]. See IE-C1, IE-C2
IE-N-28	See also Requirement IE-C17 for requirements for rare and extremely rare events. See IE-C1, IE-C2
IE-N-29	See [IE-1] for a source of industry experience. See IE-C6
IE-N-30	In the case of a plant composed of one reactor, estimating events on a per plant calendar year is equivalent to events per reactor calendar year. For plants composed of two or more reactors, each initiating event is estimated on a per plant calendar year basis so the results for the quantification of event sequences is always on a per plant calendar year basis. It is necessary to identify whether each event impacts each reactor independently or impacts a combination of reactors to perform the calculation. When determining total risk, which includes contributions from events occurring during power operation as well as during other plant operating states, the calculation of the contribution for each plant operating state must account for the fraction of the year that the plant is in that plant operating state. In this standard, for PRAs on single or multiple reactors, the appropriate units for the initiating event frequency are events per plant-calendar-year, commonly expressed as events per plant-year. The term "plant" in this context refers to the collection of reactors and sources of radioactive material within the scope of the PRA. Three simple examples follow: Loss of Bus Initiating Event: A loss of bus initiating event for a single reactor plant at-power plant operating state can be computed by annualizing the hourly failure rate of the bus and the associated breakers, relays, etc., that could lead to loss of power on the bus during the time the plant is at-power. For example, for the bus itself, the initiating event frequency over a full year would be calculated as follows: $f_{bus-8760} = \lambda_{bus} H_{year}$, where: $f_{bus-8760}$ = frequency of loss of bus over a full 8760 hr./yr.;

Table IE-1 Notes Supporting Initiating Event Analysis Requirements (Cont'd)

Number	Notes
IE-N-30 (Cont'd)	λ_{bus} = failure rate of bus per hour, say, 10^{-7}/hr.; H_{year} = hours in 1 calendar-yr. or plant-yr., 8760 hr./yr. However, to calculate each event sequence frequency for events at-power only (i.e., for the scope of PRA covered by this Standard), it is necessary to adjust for the fraction of time the plant is at-power. Thus, the result obtained from the above equation needs to be multiplied by an additional term, say, $F_{at-power}$, $F_{at-power}$ = fraction of year that, on average, the plant is at-power, for example, 90%. Thus, $f_{busat-power}$ = 10^{-7}/hr. * 8760 hr./yr. * 0.90 = 7.9×10^{-4}/plant-calendar-yr. Turbine Trip Initiating Event: Some initiating events, such as a turbine trip initiating event, may be computed based on plant-specific experience. In this case, the number of events classified as turbine trip events is in the numerator, and the number of applicable calendar-years of operation is in the denominator. The fraction of time at-power is implicitly included in the numerator because the turbine trip experience is limited to at-power experience by the nature of the event. Thus, for the same plant comprised of one reactor: $f_{TT} = N_{TT}/Y_{OP}$, f_{TT} = frequency of turbine trip events per plant-calendar-year; N_{TT} = number of events classified as turbine trip events, for example, 27 events; Y_{OP} = number of applicable calendar-years of plant operation (regardless of operating mode), for example, 23 yr.; f_{TT} = 27 events/23 yr = 1.2/plant-calendar-yr. For each of the above examples, the numbers would work out to be the same if the events were calculated on a reactor-calendar-year basis because there is only one reactor in the plant. For a PRA on a two-reactor plant, an initiating event that impacts both reactors concurrently, such as loss of off-site power (LOOP), is measured in units of events per plant-calendar-year. For an initiating event that impacts only a single reactor, the frequency per plant-calendar-year would be twice the frequency measured on a reactor-calendar-year basis when compared to the case for a single reactor PRA. The number of applicable calendar-years should be based on the time period of the event data being used and may exclude unusual periods of nonoperation (i.e., if the plant was in an extended forced shutdown). For some applications, such as configuration risk management or analyses that compare specific risks during different modes of operation, it may be appropriate to utilize initiating event frequencies that do not consider the fraction of time in the operating state. In these cases, the initiating event frequency should simply be per unit of time (i.e., per hour or per year). For at-power operation, this basis is sometimes referred to as per plant-critical-year (i.e., assuming that the reactor plant operated continuously for 1 yr). On a more general basis, it could be considered to be per plant-operating-state-year. In the loss-of-bus-initiating event example above, the term $F_{at-power}$ would not be included in the computation of initiating event frequency for these kinds of applications. In the turbine-trip-initiating event example above, the value must be adjusted by dividing f_{TT} by $F_{at-power}$. Refer to [IE-4] and [IE-5] for additional guidance on modeling multi-reactor initiating events. See IE-C8
IE-N-31	Operational performance is not expected for pre-operational plants. See IE-C10
IE-N-32	When fault trees are used to quantify support system initiating events, it is important to account for whether the failure being represented will occur during the plant operating state under consideration. A procedural event tree, where the top events represent operator actions within the governing outage procedures (Requirement IE-C4), may also be used. See IE-C11

Table IE-1 Notes Supporting Initiating Event Analysis Requirements (Cont'd)

Number	Notes
IE-N-33	One example of modified Human Reliability Analysis calculation method for at-initiator is described as follows: *(a)* estimate the human error probabilities (HEPs) conditional on the occurrence of the activity using a systematic process consistent with the SRs of HLR-HR-F, HLR-HR-G, and HLR-HR-H, as appropriate, or using expert judgment (see Section 4.2); *(b)* estimate the frequency of the associated activity. See IE-C12
IE-N-34	Examples of approaches to modeling contribution of operators to an initiating event frequency include: *(a)* an HFE may be modeled explicitly as causing an initiating event such as a line-up error on the operating train that was directed by the control room, so the contribution of operators in the control room is 100%; *(b)* alternatively, an initiating event may involve a maintenance alignment error but it was not directed by the control room, so the contribution of operators in control room is 0%; *(c)* support system initiating event fault tree analysis. See IE-C13
IE-N-35	This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding. See IE-C19

4.3.3 Event Sequence Analysis (ES)

This Section presents the technical requirements associated with Event Sequence Analysis.

4.3.3.1 Objectives and Technical Requirements for Event Sequence Analysis

The objectives of the Event Sequence Analysis ensure that

(a) the sources of radioactive material, the barriers to radionuclide release, and the safety functions necessary to protect each barrier for each source within the scope of the PRA model are defined as a basis for the event sequence model development and described for each plant operating state;

(b) plant-, design- and site-specific dependencies that impact significant event sequences are represented in the event sequence structure;

(c) individual function successes, mission times, and time windows for operator actions for each reactor-specific safety function and release phenomenon modeled in the event sequences are accounted for; and

(d) the Event Sequence Analysis is documented to provide traceability of the work.

Table 4.3.3.1-1 High Level Requirements for Event Sequence Analysis

Designator	Requirement
HLR-ES-A	The Event Sequence Analysis shall describe the plant- or design-specific scenarios that can lead to the release of radioactive material following each modeled initiating event for all modeled plant operating states and sources of radioactive material consistent within the scope of the PRA. These scenarios shall address system responses and operator actions, including recovery actions that support the key safety functions necessary to protect radionuclide barriers and to prevent or mitigate the release of radioactive material.
HLR-ES-B	Dependencies that can impact the ability of the radionuclide barriers, mitigating systems, and operator actions to operate and function shall be addressed.
HLR-ES-C	The Event Sequence Analysis shall account for a necessary and sufficient set of system and radionuclide transport barrier responses, operator actions, and release phenomena to be able to support the unambiguous definition of the reactor-specific release categories and to determine the associated mechanistic source terms. The end states of the event sequence development shall include successful prevention of the release of radioactive material.
HLR-ES-D	The documentation of the Event Sequence Analysis shall provide traceability of the work.

Table 4.3.3.1-2 Supporting Requirements for HLR-ES-A

The Event Sequence Analysis shall describe the plant- or design-specific scenarios that can lead to the release of radioactive material following each modeled initiating event for all modeled plant operating states and sources of radioactive material consistent within the scope of the PRA. These scenarios shall address system responses and operator actions, including recovery actions that support the key safety functions necessary to protect radionuclide barriers and to prevent or mitigate the release of radioactive material. (HLR-ES-A)

Index No. ES-A	Capability Category I	Capability Category II
ES-A1	USE a method for Event Sequence Analysis that *(a)* explicitly models the appropriate combinations of system responses and operator actions that affect the key safety functions, for each modeled initiating event plant evolution, and plant operating state, and changing plant conditions within a plant operating state; *(b)* includes a clear definition of event sequences; *(c)* provides a framework to support Event Sequence Quantification. See Note ES-N-1	
ES-A2	For each source of radioactive material selected within the scope of the PRA model, IDENTIFY the radionuclide transport barriers to the release from the plant. See Note ES-N-2, ES-N-3	
ES-A3	For each modeled initiating event and its source(s) of radioactive material, IDENTIFY the key reactor-specific safety functions that are necessary to protect its radionuclide transport barriers and reach a safe, stable end state and prevent or mitigate a release of radioactive material. See Note ES-N-2	

Table 4.3.3.1-2 Supporting Requirements for HLR-ES-A (Cont'd)

The Event Sequence Analysis shall describe the plant- or design-specific scenarios that can lead to the release of radioactive material following each modeled initiating event for all modeled plant operating states and sources of radioactive material consistent within the scope of the PRA. These scenarios shall address system responses and operator actions, including recovery actions that support the key safety functions necessary to protect radionuclide barriers and to prevent or mitigate the release of radioactive material. (HLR-ES-A)

Index No. ES-A	Capability Category I	Capability Category II
ES-A4	For each modeled initiating event, using the success criteria (in accordance with Requirement SC-A5) defined for each key safety function identified in Requirement ES-A3, IDENTIFY the necessary operator actions to achieve the defined success criteria. See Note ES-N-2, ES-N-4	
ES-A5	DEVELOP the event sequence model in a manner that is consistent with the plant-specific or design-specific, as applicable, transient response in performance of safety functions associated with the prevention and mitigation of a release of each modeled source of radioactive material in each modeled plant operating state. INCLUDE operator actions based on emergency operating procedures, abnormal procedures, shutdown operating procedures, operator training and guidance, event management guidelines, and technical support center guidance when available.	
ES-A6	Where practical, sequentially ORDER the events representing the response of the systems, barriers, and operator actions according to the timing of the event as it occurs in the event sequence progression. Where not practical, PROVIDE the rationale used for the ordering.	
ES-A7	DELINEATE the possible event sequences for each modeled initiating event, unless the sequences can be demonstrated to be a non-risk-significant contributor. RETAIN any sequences that do not meet SCR-3 in Table 1.10-1.	
ES-A8	DEFINE the end state of the event sequence as occurring when either a release of radioactive material in one of the reactor-specific release categories is expected to occur or a steady-state condition has been reached in which each safety function is fulfilled and a radioactive release above the level defined in RI-A5 has been prevented.	
ES-A9	INCLUDE in the event sequence and end state definition the number and specific combination of reactors and sources of radioactive material involved in the event sequence definition and release of radioactive material.	
ES-A10	USE generic plant response analyses (e.g., as performed by a plant vendor for a class of similar plants) to determine the event sequence parameters that could potentially affect the operability of the mitigating systems. JUSTIFY applicability of reference analysis. See Note ES-N-5	USE realistic, applicable (i.e., design-specific or from similar plants) thermal-hydraulic analyses to determine the event progression parameters (e.g., timing, temperature, pressure, steam) that could potentially affect the operability of the mitigating systems. See Requirement SC-B4 See Note ES-N-5
ES-A11	PERFORM the plant response analysis at a level of detail that is consistent with that of the design and operational information sufficient to identify potential risk-significant contributors.	
ES-A12	In constructing the event sequence models, INCLUDE, for each modeled initiating event, individual events in the event sequence sufficient to bound system operation, timing, and operator actions necessary for key safety functions.	In constructing the event sequence models, INCLUDE, for each modeled initiating event, sufficient detail that risk-significant differences in requirements on systems and operator responses (e.g., systems initiations or valve alignment) are included. Where diverse systems and/or operator actions provide a similar function, if choosing one over another, change the requirements for operator intervention or the need for other systems, and MODEL each separately.

Table 4.3.3.1-2 Supporting Requirements for HLR-ES-A (Cont'd)

The Event Sequence Analysis shall describe the plant- or design-specific scenarios that can lead to the release of radioactive material following each modeled initiating event for all modeled plant operating states and sources of radioactive material consistent within the scope of the PRA. These scenarios shall address system responses and operator actions, including recovery actions that support the key safety functions necessary to protect radionuclide barriers and to prevent or mitigate the release of radioactive material. (HLR-ES-A)

Index No. ES-A	Capability Category I	Capability Category II
ES-A13	Intermediate end states and/or transfers between or among two or more event trees may be used to reduce the size and complexity of individual event trees that link the initiating events to end states of the event sequence model. DEFINE any intermediate end states and transfers that are used and the method that is used to implement them in the qualitative definition of event sequences and in their quantification consistent with the Requirements SC-A2 and SC-A3. USE a method for implementing intermediate end states and event tree transfers that preserve the dependencies that are part of the transferred sequence. These include functional, system, initiating event, operator, phenomenological, and spatial or environmental dependencies.	
ES-A14	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with event sequence definition in a manner that supports the applicable requirements of HLR-ESQ-E.	
ES-A15	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built and as-operated details that influence event sequence definition. See Note ES-N-6, ES-N-7, ES-N-8	

Table 4.3.3.1-3 Supporting Requirements for HLR-ES-B

Dependencies that can impact the ability of the radionuclide barriers, mitigating systems, and operator actions to operate and function shall be addressed. (HLR-ES-B)

Index No. ES-B	Capability Category I	Capability Category II
ES-B1	For each modeled initiating event within each modeled plant configuration and plant operating state, IDENTIFY mitigating systems and radionuclide transport barriers that are challenged, degraded, or failed by the occurrence of the initiator. INCLUDE the impact of initiating events on radionuclide transport barriers and mitigating systems in the event sequence either in the event sequence models or in the system models. See Note ES-N-9	
ES-B2	IDENTIFY the dependence of modeled mitigating systems on the success or failure of preceding systems, functions, and human actions. INCLUDE the impact on event sequence definition, either in the event sequence models or in the system models.	
ES-B3	For each event sequence, IDENTIFY the phenomenological conditions created by the event sequence including those caused by changes in plant configurations during the modeled plant operating state. Phenomenological impacts include generation of harsh environments affecting temperature, pressure, debris, water levels, humidity, etc., that could impact the success of the system, function, chemical reactions, formation of combustible materials, or fires involving key reactor components exposed to harsh environments, or operator action under consideration. INCLUDE the impact of the event sequence phenomena, either in the event sequence models or in the system models. See Note ES-N-10	
ES-B4	When the event trees with the conditional split fraction method is used, if the probability of Event B is dependent on the occurrence or nonoccurrence of Event A, where practical, PLACE Event A to the left of Event B in the ordering of event tops. Where not practical, DESCRIBE the rationale used for the ordering.	

Table 4.3.3.1-3 Supporting Requirements for HLR-ES-B (Cont'd)

Dependencies that can impact the ability of the radionuclide barriers, mitigating systems, and operator actions to operate and function shall be addressed. (HLR-ES-B)

Index No. ES-B	Capability Category I	Capability Category II
ES-B5	DEVELOP the event sequence models to a level of detail sufficient to identify intersystem dependencies and train level interfaces, either in the event trees or through a combination of event tree and fault tree models and associated logic.	
ES-B6	ENSURE that the level of detail in delineating the dependencies is consistent with the level of detail of the design information and procedural guidance available sufficient to identify potential risk-significant contributors. If dependency information is missing in the supporting design information, IDENTIFY assumptions regarding dependence and independence among SSCs, barriers, and operator actions.	
ES-B7	If plant configurations and maintenance practices create or alter dependencies among various systems, DEFINE and MODEL these configurations and alignments in a manner that represents these dependencies, either in the event sequence models or in the system models.	
ES-B8	MODEL time-phased dependencies (i.e., those that change as the event progresses, due to such factors as depletion of resources, recovery of resources, and changes in loads) in the event sequences. See Note ES-N-11	
ES-B9	IDENTIFY the sources of model uncertainty, related assumptions, and reasonable alternatives in the event sequence dependency analysis in a manner that supports the applicable requirements of HLR-ESQ-E.	
ES-B10	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built and as-operated details that influence event sequence dependency analysis. See Note ES-N-7, ES-N-8	

Table 4.3.3.1-4 Supporting Requirements for HLR-ES-C

The Event Sequence Analysis shall account for a necessary and sufficient set of system and radionuclide transport barrier responses, operator actions, and release phenomena to be able to support the unambiguous definition of the reactor-specific release categories and to determine the associated mechanistic source terms. The end states of the event sequence development shall include successful prevention of the release of radioactive material. (HLR-ES-C)

Index No. ES-C	Capability Category I	Capability Category II
ES-C1	DEFINE the end states of the event sequence model in sufficient detail to resolve different event sequence families. INCLUDE safe stable end states that prevent the release of radioactive material and release categories with different mechanistic source terms in accordance with HLR-MS-A. See Note ES-N-12	
ES-C2	In the definition of release categories, IDENTIFY those physical characteristics at the time of release or the time of reaching an intermediate end state, if used, that can influence the mechanistic source term from the sources of radioactive material within the scope of the PRA. If the event sequence model employs intermediate states or plant damage states (PDSs) prior to the resolution of release categories, IDENTIFY the associated physical characteristics at the time that the intermediate states are reached. See Note ES-N-13, ES-N-14	
ES-C3	IDENTIFY the event sequence characteristics that lead to the physical characteristics identified in Requirement ES-C2. See Note ES-N-3, ES-N-15	

Table 4.3.3.1-4 Supporting Requirements for HLR-ES-C (Cont'd)

The Event Sequence Analysis shall account for a necessary and sufficient set of system and radionuclide transport barrier responses, operator actions, and release phenomena to be able to support the unambiguous definition of the reactor-specific release categories and to determine the associated mechanistic source terms. The end states of the event sequence development shall include successful prevention of the release of radioactive material. (HLR-ES-C)

Index No. ES-C	Capability Category I	Capability Category II
ES-C4	IDENTIFY how the physical characteristics identified in Requirement ES-C2 and the event sequence characteristics identified in Requirement ES-C3 are addressed in the Event Sequence Analysis. JUSTIFY excluding from the Event Sequence Analysis any characteristics identified in Requirement ES-C2 or in Requirement ES-C3 by demonstrating there is no significant impact on risk. See Note ES-N-16	
ES-C5	PROVIDE an event sequence development method to explicitly account for the characteristics in Requirements ES-C2 and ES-C3. These include functional, system, initiating event, operator, phenomenological, and spatial or environmental dependencies. See Note ES-N-17	
ES-C6	If the event sequence model uses PDSs to provide interfaces between different elements of the event sequence model, DEFINE such PDSs consistent with Requirements ES-C2, ES-C3, ES-C4, and ES-C5.	
ES-C7	USE supporting plant response analyses in accordance with the applicable CC-I Supporting Requirements (SRs) of HLR-SC-A and HLR-SC-B.	USE supporting plant response analyses in accordance with the applicable Capability Category II (CC-II) SRs of HLR-SC-A and HLR-SC-B.
ES-C8	DEVELOP event sequences to a level of detail consistent with available design information to account for the potential risk-significant contributors to each modeled event sequence and event sequence family identified in Requirements ES-C2 and ES-C3. ENSURE consistency in the modeling of challenges to radionuclide transport barriers in the Event Sequence Analysis with the mechanistic source term calculations. JUSTIFY any generic plant- or design-specific calculations or references used to categorize event sequences into each event sequence families.	
ES-C9	If crediting repair, ENSURE the credit is given using a conservative repair failure probability.	If crediting repair, REVIEW risk-significant event sequences resulting in each modeled event sequence family to determine if repair of equipment can be credited. JUSTIFY credit given for repair (i.e., ensure that plant conditions do not preclude repair and adequate justification exists from which to estimate the repair failure probability per Requirements SY-A31 and DA-C20). AC power recovery based on generic data applicable to the plant is acceptable.
ES-C10	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the event sequence radionuclide transport analysis in a manner that supports the applicable requirements of HLR-ESQ-E.	
ES-C11	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built and as-operated details that influence event sequence radionuclide transport analysis. See Note ES-N-7, ES-N-8	

Table 4.3.3.1-5 Supporting Requirements for HLR-ES-D

The documentation of the Event Sequence Analysis shall provide traceability of the work. (HLR-ES-D)

Index No. ES-D	Capability Category I	Capability Category II
ES-D1	DOCUMENT the process used in the Event Sequence Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the linkage between the modeled plant operating states, initiating events, and event sequences; *(b)* the success criteria established for each modeled initiating event including the bases for the criteria (i.e., the system capacities required to prevent or mitigate a release of radioactive material and the necessary components required to achieve these capacities); *(c)* each deterministic analysis performed to support the Event Sequence Analysis; *(d)* a description of the event sequence for each sequence or group of similar sequences (i.e., descriptions of the sequence timing, applicable procedural guidance, expected environmental or phenomenological impacts, dependencies between systems and operator actions, end states, and other pertinent information required to fully establish the sequence of events); this includes an evaluation of uncertainties in the Event Sequence Analysis; *(e)* the technical basis for the treatment of each of the radionuclide transport barriers during each of the modeled event sequence families. This includes the capabilities of fuel that are credited in the analysis, the structural capabilities, and the capacities of each transport barrier that is credited in the assignment of end states in the event sequence development; *(f)* the evaluation of failure modes, failure and degradation mechanisms, loading conditions, capability to withstand loads, and impacts of modeling uncertainties on radionuclide transport barrier effectiveness; *(g)* a clear definition of the event sequence end states, event sequence families, and release categories and sufficient detail in the event sequence definition to determine each event sequence family and release category with a unique mechanistic source term; *(h)* the operator actions represented in the event trees, and the sequence-specific timing and dependencies that are traceable to the Human Reliability Analysis for these actions; *(i)* the interface of the event sequence models with the release categories and mechanistic source terms as defined in the Mechanistic Source Term Analysis; *(j)* (when sequences are modeled using a single top event fault tree) the manner in which the requirements for Event Sequence Analysis has been satisfied; *(k)* mitigating systems that are challenged, degraded, or failed by each specific initiating event, and the impact on the system; *(l)* the dependence of modeled mitigating systems on the success or failure of preceding system's functions and human actions.	
ES-D2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirements ES-A14, ES-B9, and ES-C10) associated with the Event Sequence Analysis.	
ES-D3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Event Sequence Analysis. See ES-A15, ES-B10, ES-C11 See Note ES-N-8	

4.3.3.2 Peer Review Requirements for Event Sequence Analysis

4.3.3.2.1 Purpose

This Section provides requirements for peer review of the Event Sequence Analysis element of the PRA.

4.3.3.2.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the area of event sequence analysis. The team members assigned to review Event Sequence Analysis shall overlap that assigned to review Initiating Event Analysis to ensure consistency between the modeling for both elements. The team members assigned to review the Event Sequence Analysis shall have experience specific to these areas and the capability of recognizing plant-specific features of the analyses.

4.3.3.2.3 Review of Event Sequence Analysis to Confirm the Methodology

A review shall be performed on selected event sequences. The portion of the event sequences selected for review typically includes the following:

(a) definition of end states for all modeled event sequences for each plant operating state and radionuclide source within the scope of the PRA;

(b) treatment of dependencies including the use of transfers and intermediate plant states;

(c) grouping of event sequences into event sequence families and basis for similarity;

(d) event sequence models for representative non-full power plant operating state initiating events;

(e) event sequence model for a balance-of-plant transient;

(f) event sequence model containing loss of off-site power/station blackout (SBO) considerations, as applicable;

(g) event sequence model for a loss of a support system initiating event;

(h) event sequence model for reactor coolant system boundary (RCB) breaches of representative sizes and locations;

(i) event sequence model for RCB breaches bypassing containment, confinement, or reactor building, as applicable;

(j) reactor coolant system heat exchanger breaches;

(k) event sequence model for reactivity excursions and transients;

(l) treatment of reactor technology-specific phenomena in event sequence development;

(m) treatment of event sequences involving multiple reactors;

(n) treatment of event sequences involving noncore sources of radioactive material.

NONMANDATORY APPENDIX ES: NOTES AND EXPLANATORY MATERIAL FOR EVENT SEQUENCE ANALYSIS

ES.1 NOTES ASSOCIATED WITH EVENT SEQUENCE ANALYSIS

Table ES-1 Notes Supporting Event Sequence Analysis Requirements

Number	Notes
ES-N-1	Acceptable methods for clear definition of event sequences include graphical representations of event sequences such as event trees and event sequence diagrams. See ES-A1
ES-N-2	Analysis support for this assessment includes deterministic calculations using computer codes or hand calculations. See ES-A2, ES-A3, ES-A4
ES-N-3	For the reactor core source of radioactive material, the barriers typically include physical barriers such as a fuel barrier; an RCB; and a reactor building, or confinement barrier. For other sources of radioactive material, the barriers may include spent fuel pools, piping, tanks, and auxiliary buildings. In addition, there may be a process barriers that mitigate releases by providing time delays to permit radioactive decay, plateout, settling, scrubbing, and other time-dependent processes to reduce the magnitude and delay the timing of the source term. See ES-A2, ES-C3
ES-N-4	The intent of this requirement is not to address specific procedures but rather to identify, at a functional level, what operator actions are required for success. See ES-A4
ES-N-5	Examples of event sequence parameters are timing, temperature, and pressure. See ES-A10
ES-N-6	Analysis support for this identification includes deterministic calculations using computer codes or hand calculations. See SRs of HLR-SC-B for guidance on selecting appropriate computational tools. See ES-A15
ES-N-7	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See ES-A15, ES-B10, ES-C11
ES-N-8	This SR is not applicable to operating plants. See ES-A15, ES-B10, ES-C11, ES-D3
ES-N-9	The impact of initiating events on radionuclide transport barriers might include modeling dependence between an operator-induced initiating event and recovery events, especially at shutdown. See ES-B1
ES-N-10	Examples of hazards that could impact functions are loss of pump net positive suction head, clogging of flow paths, pipe whip, jet impingement, and other high-energy-line-break impacts such as flooding. See ES-B3

Table ES-1 Notes Supporting Event Sequence Analysis Requirements (Cont'd)

Number	Notes
ES-N-11	Examples of time-phased dependencies are the following: *(a)* for station blackout/loss-of-off-site-power sequences, key time-phased events, such as *(1)* alternating-current power recovery, *(2)* direct-current battery adequacy (time-dependent discharge), *(3)* environmental conditions (e.g., room cooling) for operating equipment and the control room; *(b)* for transients with failure of reactivity control, key time-dependent operator actions; *(c)* dependencies between the initiator and subsequent recovery events for event sequences initiated during non-full-power plant operating states. See ES-B8
ES-N-12	The event sequence model must include an adequate definition of release categories to define the end states of the event sequence model and to support the grouping of event sequences into event sequence families according to the definition in Section 2. There are additional requirements in HLR-MS-A to ensure that the release category definitions are refined in sufficient detail to perform the mechanistic source term calculations. See ES-C1
ES-N-13	Examples of physical characteristics include, but are not limited to, the following: *(a)* temperature and state of the radionuclide sources being modeled; *(b)* reactor coolant system (RCS) pressure (high RCS pressure can increase the magnitude of the source term and lead to challenges to radionuclide transport barriers); *(c)* status of the RCB or other relevant barriers for each modeled source; *(d)* status of radionuclide transport barrier isolation (failure of isolation can result in an unscrubbed or unfiltered release); *(e)* status of any radionuclide transport barrier filtration systems; *(f)* radionuclide transport barrier integrity (e.g., vented, bypassed, or failed); *(g)* number of reactors or sources of radioactive material involved in the release; *(h)* time of physical events after the initiating event (e.g., trip or shutdown). See ES-C2
ES-N-14	Examples of sources of radioactive material besides that in the reactor core that may be included in the scope of the PRA include radioactive material circulating or initially plated out within the RCB that may be lifted off during a depressurization event, spent fuel in the spent fuel storage system, fuel/salt systems outside the reactor core, offgas systems, radioactive waste systems, and other process systems with radioactive material. See ES-C2

Table ES-1 Notes Supporting Event Sequence Analysis Requirements (Cont'd)

Number	Notes
ES-N-15	Examples of event sequence characteristics that lead to the physical characteristics include but are not limited to the following: *(a)* reactors and/or non-reactor sources of radioactive material whose event sequences are being modeled; *(b)* type of initiator *(1)* transients that can result in high RCS pressure, *(2)* RCB breaches that may result in lower RCS pressure, *(3)* RCB breaches that result in radionuclide transport barrier bypass, *(4)* vents that pose a unique challenge to any radionuclide transport barrier such as those associated with intrinsic hazards; *(c)* status of dependencies and support systems (e.g., electric power); loss of a support system can result in unavailability of the supported mitigating systems; *(d)* status of each of the radionuclide transport barriers for each source involved in the release; *(e)* status of radionuclide transport barriers; *(f)* status of mitigating systems including availability and accessibility of mitigating equipment that may influence the magnitude of the radioactive release source term from the sources of radioactive material within the scope of the PRA; *(g)* status of support systems [e.g., electrical power, component cooling, and heating/ventilation and air conditioning (HVAC)]; *(h)* environmental or physical conditions introduced by the hazard, if any, that may interfere with recovery actions that would occur after the onset of radionuclide release; *(i)* design and physical configuration of reactor coolant system, primary and secondary radionuclide transport barriers, and other neighboring structures, if treated; *(j)* adverse effects of intrinsic hazards on barriers and safety functions; *(k)* physical effects of the flooding of plant areas and structures, systems, and components (SSCs) serving as radionuclide transport barriers and/or auxiliary building(s) on radionuclide release or event sequence mitigation (e.g., submergence of release pathway or impeding human actions); *(l)* status of other reactors and radionuclide sources in the plant and status of shared systems between reactors. See ES-C3
ES-N-16	Consider physical and event sequence characteristics that are addressed in the following: *(a)* a single-stage event tree model; *(b)* a multistage event tree model; *(c)* bridge trees; and *(d)* each of any linked event trees. See ES-C4
ES-N-17	Example methods include the use of a single event tree or series of linked event trees between the initiating event and event sequence family end states, construction of a bridge tree, transfer of the information via PDSs, or a combination of these. See ES-C5

4.3.4 Success Criteria Development (SC)

This Section presents the technical requirements associated with Success Criteria Development.

4.3.4.1 Objectives and Technical Requirements for Success Criteria Development

The objectives of the Success Criteria Development ensure that

(a) overall success criteria are defined (i.e., prevention of a release of radioactive material) in each of the modeled event sequences and event sequence families as defined in the Event Sequence Analysis element and release categories as defined in the Mechanistic Source Term Analysis element;

(b) success criteria are defined for key safety functions, supporting systems, structures, barriers to release of radioactive material, components, and operator actions necessary to support event sequence development based on a defensible technical basis; and

(c) the success criteria is documented to provide traceability of the work.

NOTE: For PRA models using multivalued (nonbinary) logic, the term "success criteria" in the following High Level Requirements (HLRs) and Supporting Requirement (SRs) should be interpreted as the criteria used to determine the different event outcomes.

Table 4.3.4.1-1 High Level Requirements for Success Criteria Development

Designator	Requirement
HLR-SC-A	The overall success criteria for the system, structure, component, and human action success criteria used in the PRA shall be defined and referenced and shall be consistent with the reactor safety design approach, features, available procedures, and design and/or operating philosophy of the plant. This includes defining end states, establishing event sequence mission times, and ensuring that mitigating systems shared between reactors are addressed.
HLR-SC-B	The thermal fluid, structural, and other supporting engineering bases shall be capable of providing success criteria and event timing sufficient for quantification of each modeled event sequence; any modeled intermediate end state and event sequence family frequency; determination of the relative impact of success criteria on structures, systems, components, and human actions; and the impact of uncertainty.
HLR-SC-C	The documentation of the Success Criteria Development shall provide traceability of the work.

Table 4.3.4.1-2 Supporting Requirements for HLR-SC-A

The overall success criteria for the system, structure, component, and human action success criteria used in the PRA shall be defined and referenced and shall be consistent with the reactor safety design approach, features, available procedures, and design and/or operating philosophy of the plant. This includes defining end states, establishing event sequence mission times, and ensuring that mitigating systems shared between reactors are addressed. (HLR-SC-A)

Index No. SC-A	Capability Category I	Capability Category II
SC-A1	DEFINE a safe stable state and SPECIFY the basis for the selected definition. See Note SC-N-1	
SC-A2	In the definition of end states for event sequences and event sequence families in the Event Sequence Analysis, USE the definition of reactor-specific release categories whose requirements are defined in the Mechanistic Source Term Analysis element. If intermediate end states such as plant damage states are used to define end states in the Event Sequence Analysis, SPECIFY the physical plant conditions associated with such end states and the corresponding success criteria.	

Table 4.3.4.1-2 Supporting Requirements for HLR-SC-A (Cont'd)

The overall success criteria for the system, structure, component, and human action success criteria used in the PRA shall be defined and referenced and shall be consistent with the reactor safety design approach, features, available procedures, and design and/or operating philosophy of the plant. This includes defining end states, establishing event sequence mission times, and ensuring that mitigating systems shared between reactors are addressed. (HLR-SC-A)

Index No. SC-A	Capability Category I	Capability Category II
SC-A3	SPECIFY the plant parameters and associated criteria (e.g., temperature limits) to be used in determining the end states used in the Event Sequence Analysis. SPECIFY whether a release of radioactive material is below the value defined in Requirement RI-A5, and if above this value, which of the modeled release categories would result.	SPECIFY the plant parameters and associated criteria (e.g., temperature limits) to be used in determining the end states used in the Event Sequence Analysis. Select these parameters such that the determination of the end state is as realistic as practical, consistent with current best practice. SPECIFY whether a release of radioactive material is below the value defined in Requirement RI-A5, and if above this value, which of the modeled release categories would result. SPECIFY criteria with sufficient margin on the code-calculated or empirically established values to allow for limitations of the code, sophistication of the models, and uncertainties in the results and applied empirical data, consistent with the requirements specified under the SRs of HLR-SC-B.
SC-A4	SPECIFY the plant parameters and associated criteria for protecting the integrity of each radionuclide transport barrier.	
SC-A5	SPECIFY success criteria for each of the key safety functions identified per Requirement ES-A3 for each modeled initiating event and associated plant operating state. See Note SC-N-2	
SC-A6	IDENTIFY mitigating systems that are shared between reactors, between non-core sources of radioactive material, and between reactors and non-core sources of radioactive material and the manner in which the sharing is performed in the event that they share a common initiating event or failure mode [e.g., loss of off-site power (LOOP)].	

Table 4.3.4.1-2 Supporting Requirements for HLR-SC-A (Cont'd)

The overall success criteria for the system, structure, component, and human action success criteria used in the PRA shall be defined and referenced and shall be consistent with the reactor safety design approach, features, available procedures, and design and/or operating philosophy of the plant. This includes defining end states, establishing event sequence mission times, and ensuring that mitigating systems shared between reactors are addressed. (HLR-SC-A)

Index No. SC-A	Capability Category I	Capability Category II
SC-A7	SPECIFY a sequence mission time for the modeled event sequences to achieve a safe stable state. For sequences in which a safe stable state has been achieved, USE a minimum sequence mission time of 24 hours. For sequences in which a safe stable state would not be achieved within 24 hours using the modeled plant equipment and human actions, ASSIGN a conservative end state.	SPECIFY a sequence mission time for the modeled event sequences to achieve a safe stable state. For sequences in which a safe stable state has been achieved, USE a minimum sequence mission time of 24 hours. For sequences in which a safe stable state would not be achieved within 24 hours using the modeled plant equipment and human actions, PERFORM additional evaluation or modeling by using techniques such as *(a)* assigning an event sequence end state for the event sequence; *(b)* extending the sequence mission time and adjusting the affected analyses to the point at which a safe stable state can be demonstrated; or *(c)* modeling additional system recovery or operator actions for the sequence, in accordance with requirements stated in Systems Analysis and Human Reliability Analysis, to demonstrate that a successful outcome is achieved.
SC-A8	ENSURE the component mission time supports the full sequence mission time for which the component is credited or JUSTIFY a shorter component mission time. See Note SC-N-3	
SC-A9	ENSURE that the bases for the success criteria are consistent with the features, available procedures, and design and/or operating philosophy of the plant.	
SC-A10	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with success criteria in a manner that supports the applicable requirements of HLR-ESQ-E.	
SC-A11	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the success criteria. See Note SC-N-4, SC-N-5	

Table 4.3.4.1-3 Supporting Requirements for HLR-SC-B

The thermal fluid, structural, and other supporting engineering bases shall be capable of providing success criteria and event timing sufficient for quantification of each modeled event sequence; any modeled intermediate end state and event sequence family frequency; determination of the relative impact of success criteria on structures, systems, components, and human actions; and the impact of uncertainty. (HLR-SC-B)

Index No. SC-B	Capability Category I	Capability Category II
SC-B1	DEFINE success criteria by using generic analyses that are applicable to the type of reactor and plant design. See Note SC-N-6	DEFINE the realistic success criteria to mitigate the risk-significant event sequences based on applicable generic and plant- or design-specific analyses. For non-risk-significant event sequences, ENSURE the requirement of Capability Category I (CC-I) is met. See Note SC-N-6
SC-B2	SATISFY the requirements of Section 4.2, Use of Expert Judgment, in those situations in which there is a lack of empirical information regarding the condition or response of a modeled structure, system, and component (SSC), or a lack of analytical methods upon which to base a prediction of SSC condition or response.	
SC-B3	When defining success criteria, USE thermal fluid, structural, or other analyses consistent with the level of detail of the initiating event grouping (HLR-IE-B), plant operating state definition (HLR-POS-A), and event sequence modeling (Requirements in HLR-ES-A and HLR-ES-B). See Note SC-N-7	
SC-B4	USE *(a)* analysis models, validated computer codes, and data that have sufficient capability to model or represent the conditions of interest in the determination of success criteria for end states and that provide results representative of the plant; *(b)* an accepted process for verification and validation of computer programs, or JUSTIFY an alternative method; *(c)* computer codes, data, and models only within known limits of applicability. See Note SC-N-8, SC-N-9	
SC-B5	For defining success criteria for safety functions performed via purely passive means (i.e., relying on natural physical processes such as natural convection, thermal conduction, radiation, etc.), USE mechanistic models supported by empirical data, and CHARACTERIZE uncertainties of the applied models and input data to demonstrate that success criteria have been adequately fulfilled in the calculation of passive functional reliability. See Note SC-N-10	
SC-B6	DEFINE the thermal-mechanical loads or other relevant physical attributes of the event sequence-specific challenges to radionuclide transport barrier integrity. IDENTIFY the relevant parameters that are needed to define the capability of each radionuclide transport barrier to withstand these challenges. See Note SC-N-11	
SC-B7	SELECT a conservative method for evaluating effectiveness of each radionuclide transport barrier consistent with the definition of loads and capacities in Requirement SC-B6.	SELECT a realistic method for evaluating effectiveness of each radionuclide transport barrier. INCLUDE an assessment of the uncertainties in the event sequence-specific loading conditions of each radionuclide transport barrier and radionuclide transport barrier capacity to withstand the loads consistent with the definition of loads and capacities in Requirement SC-B6.
SC-B8	ENSURE the reasonableness and acceptability of the results of the thermal fluid, structural, or other supporting engineering bases used to support the success criteria. See Note SC-N-12	
SC-B9	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with thermal fluid, structural analyses, and other engineering bases used to develop the Success Criteria engineering basis in a manner that supports the applicable requirements of HLR-ESQ-E.	
SC-B10	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the thermal-fluid, structural analyses, and other engineering bases. See Note SC-N-4, SC-N-5	

Table 4.3.4.1-4 Supporting Requirements for HLR-SC-C

The documentation of the Success Criteria Development shall provide traceability of the work. (HLR-SC-C)

Index No. SC-C	Capability Category I	Capability Category II
SC-C1	DOCUMENT the process used in the Success Criteria Development specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the definition of end states used in the PRA (i.e., prevention of release and each modeled plant operating state, event sequence, and event sequence family), including the bases for any selected parameter value used in the definition (e.g., peak fuel temperature or reactor coolant system parameters); *(b)* calculations (generic and plant- or design-specific), empirical data, or other references used to establish success criteria, including evaluation of radionuclide transport barrier capability and effectiveness, and identification of cases for which they are used; *(c)* identification of computer codes, empirical data, or other methods used to establish plant- or design-specific success criteria and evaluation of radionuclide transport barrier capability and effectiveness; *(d)* a description of the limitations (e.g., potential conservatisms or limitations that could challenge the applicability of computer models or data in certain cases) of the calculations or codes, including the following: *(1)* uncertainties in the calculations for purely passive safety function success criteria, and *(2)* the evaluation of radionuclide transport barrier capability and effectiveness; *(e)* the uses of expert judgment within the PRA and rationale for such uses; *(f)* a summary of success criteria and the supporting technical bases for the available mitigating systems and human actions for each initiating event group modeled in the PRA; *(g)* the basis for establishing the time available for human actions; *(h)* descriptions of processes used to define success criteria for grouped initiating events or event sequences; *(i)* technical basis for success criteria associated with digital instrumentation and control systems; *(j)* technical basis for success criteria for systems that utilize passive means to perform safety functions and an identification of the uncertainties in these success criteria and how they were addressed in success criteria formulation; *(k)* mitigating systems that are shared between reactors, non-core sources of radioactive materials, or a combination of reactors and non-core sources of radioactive materials and the manner in which the sharing is performed should multiple reactors, non-core sources of radioactive materials, or a combination of reactors and non-core sources of radioactive materials experience a common initiating event, if applicable.	
SC-C2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirements SC-A10 and SC-B9) associated with the Success Criteria Development.	
SC-C3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Success Criteria Development. See SC-A11, SC-B10 See Note SC-N-5	

4.3.4.2 Peer Review Requirements for Success Criteria Development

4.3.4.2.1 Purpose

This Section provides requirements for peer review of the Success Criteria Development element of the PRA.

4.3.4.2.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the area of success criteria analysis. The team members assigned to review Success Criteria Development shall overlap that assigned to review mechanistic source term and Event Sequence Analysis to ensure consistency between the modeling for those elements. The team members assigned to review the Success Criteria Development shall have experience specific to these areas and the capability of recognizing plant-specific features of the analyses.

4.3.4.2.3 Review of Success Criteria Development to Confirm the Methodology

A review shall be performed on selected success criteria analyses. The portion of the analyses selected for review typically includes the following:

(a) the definition of event sequence end states used in the success criteria evaluations and the supporting bases;

(b) the conditions corresponding to a safe, stable state;

(c) the reactor and barrier response conditions used in defining end states and supporting bases;

(d) the radionuclide barrier success criteria used in the PRA for mitigating each modeled initiating event;

(e) the generic bases (including assumptions) used to establish the success criteria of systems credited in the PRA and the applicability to the modeled plant or design;

(f) the plant-specific bases (including assumptions) used to establish the system success criteria of systems credited in the PRA;

(g) calculations performed specifically for the PRA for each computer code used to establish safety function success criteria and event sequence timing;

(h) calculations performed specifically for the PRA for each computer code used to establish support system success criteria [e.g., a room heat-up calculation used to establish room cooling requirements or a load shedding evaluation used to determine battery life during an station blackout (SBO)];

(i) expert judgments used in establishing success criteria used in the PRA .

4.3.4.3 References for Success Criteria Development

The following is the publication referenced in this Standard.

[SC-1] NUREG/BR-0167, “Software Quality Assurance Program and Guidelines,” U.S. Nuclear Regulatory Commission, 1993

NONMANDATORY APPENDIX SC: NOTES AND EXPLANATORY MATERIAL FOR SUCCESS CRITERIA DEVELOPMENT

SC.1 NOTES ASSOCIATED WITH SUCCESS CRITERIA DEVELOPMENT

Table SC-1 Notes Supporting Success Criteria Development Requirements

Number	Notes
SC-N-1	It is anticipated that the definition and basis for a safe stable state will *(a)* be consistent with the definition in Section 2.2; and *(b)* have consequences less than the reporting threshold from RI-A5. See SC-A1
SC-N-2	Requirements for specification of success criteria appear under HLRs for other elements as well (e.g., requirements in HLR-ES-A and HLR-SY-A). These requirements are intended to be complementary, not duplicative. For example, for event sequences, Requirements SC-A5, SC-A6 (if applicable), ES-A2, ES-A3, and ES-A4 are intended to be used together to capture the specification of the set of systems and human actions necessary to meet the key safety function success criteria. See SC-A5
SC-N-3	Component mission times for individual SSCs that function during the sequence may be shorter than the sequence mission time, as long as a set of SSCs and operator actions is modeled to support the full sequence mission time. See SC-A8
SC-N-4	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See SC-A11, SC-B10
SC-N-5	This SR is not applicable to operating plants. See SC-A11, SC-B10, SC-C3
SC-N-6	During shutdown, generic evaluation of decay heat using general shutdown power correlations is sufficient for CC-I. For Capability Category II (CC-II), outage-specific decay heat calculations may be necessary to establish system success criteria. See SC-B1
SC-N-7	Full-power success criteria are not always bounding for low-power-and-shutdown conditions. See SC-B3
SC-N-8	A qualitative evaluation of a relevant application of codes, models, or analyses that has been used for a similar class of plant (e.g., Owners Group generic studies) may be used, if justified. See SC-B4
SC-N-9	An accepted process for verification and validation of computer programs is in NUREG/BR-0167 [SC-1]. See SC-B4
SC-N-10	Not all passive systems require mechanistic models, especially those that are not purely passive. The IAEA has multiple categories of passive, and several require the initial actuation or change of state of a component, such as an explosive valve, motor operated valve, or check valve. In these cases, the reliability could be dominated by the component failure rate. See SC-B5
SC-N-11	For early pre-operational stage PRAs, the scope may be limited to identifying challenges to a single barrier. See SC-B6
SC-N-12	Examples of methods to achieve this include the following: *(a)* comparison with results of the same analyses performed for similar plants, addressing differences in unique plant features; *(b)* comparison with results of similar analyses performed with other plant- or design-specific codes; *(c)* confirmation by other means that have been determined to be appropriate for an analysis. See SC-B8

4.3.5 Systems Analysis (SY)

This Section presents the technical requirements associated with Systems Analysis.

4.3.5.1 Objectives and Technical Requirements for Systems Analysis

The objectives of Systems Analysis ensure that

(a) there is a reasonably complete set of the independent system failure and unavailability modes and associated human failure events, (HFEs) and system alignments for each system;

(b) there is a reasonably complete identification of the common cause failures (CCFs) and dependency effects on system performance; and

(c) the Systems Analysis is documented to provide traceability of the work.

Table 4.3.5.1-1 High Level Requirements for Systems Analysis

Designator	Requirement
HLR-SY-A	System logic models shall be developed that represent the various system alignments, success criteria, and mission times and include the failure modes associated with system maintenance, component actuation and functionality, and associated HFEs.
HLR-SY-B	CCFs and both intersystem and intrasystem dependencies that could influence system performance shall be evaluated, and shall be modeled as applicable, including evaluating functional, human, and phenomenological effects, as well as dependencies on plant operating states.
HLR-SY-C	The documentation of the Systems Analysis shall provide traceability of the work.

Table 4.3.5.1-2 Supporting Requirements for HLR-SY-A

System logic models shall be developed that represent the various system alignments, success criteria, and mission times and include the failure modes associated with system maintenance, component actuation and functionality, and associated HFEs. (HLR-SY-A)

Index No. SY-A	Capability Category I	Capability Category II
SY-A1	IDENTIFY those systems needed to provide or support the safety functions contained in the Event Sequence Analysis for the sources of radioactive material, plant operating states, and hazard groups within the scope of the PRA. See Note SY-N-1	
SY-A2	COLLECT pertinent information to ensure that the Systems Analysis represents either the as-built, as-operated or the as-designed, as-intended-to-operate systems, as applicable. See Note SY-N-2, SY-N-3	
SY-A3	REVIEW plant information sources to define or establish the following: *(a)* system components and boundaries; *(b)* dependencies on other systems; *(c)* instrumentation and control (I&C) and associated software requirements; *(d)* testing and maintenance requirements and practices; *(e)* operating limitations such as those imposed by Technical Specifications; *(f)* component operability and design limits; *(g)* procedures for the operation of the system during normal and off-normal conditions; *(h)* system configuration during normal and off-normal conditions; *(i)* for low power and shutdown (LPSD) states, past outages should be reviewed to determine unique system operating states (e.g., temporary power or cooling) that should be included in the sequence models. If past outage data is unavailable, review design information for plant modes and states, and plant provisions for maintenance and in-service inspection to determine unique system operating states to be modeled. Meet Requirement SY-A4, if applicable.	
SY-A4	For PRAs performed during the pre-operational stage, USE available plant information sources to define or establish items *(a)* through *(i)* in SY-A3. SPECIFY and JUSTIFY any other information sources used in lieu of unavailable plant information. See Note SY-N-4	

Table 4.3.5.1-2 Supporting Requirements for HLR-SY-A (Cont'd)

System logic models shall be developed that represent the various system alignments, success criteria, and mission times and include the failure modes associated with system maintenance, component actuation and functionality, and associated HFEs. (HLR-SY-A)

Index No. SY-A	Capability Category I	Capability Category II
SY-A5	For operating reactors, CONFIRM that the Systems Analysis correctly represents the as-built, as-operated plant through discussions with knowledgeable plant personnel (e.g., engineering, plant operations). See Note SY-N-5, SY-N-6	For operating reactors, CONFIRM that the Systems Analysis correctly represents the as-built, as-operated plant, as applicable by performing plant investigation(s) and interview(s) with knowledgeable plant personnel (e.g., engineering, plant operations). See Note SY-N-5, SY-N-6
SY-A6	For PRAs performed during the pre-operational stage, CONFIRM that the Systems Analysis correctly represents the as-designed, as-intended-to-operate plant through discussions with knowledgeable plant personnel (e.g., engineering, plant operations). See Note SY-N-4, SY-N-5	For PRAs performed during the pre-operational stage, CONFIRM that the Systems Analysis correctly represents the as-designed, as-intended-to-operate plant by performing interviews and detailed investigation(s) with knowledgeable engineering personnel. See Note SY-N-4, SY-N-5
SY-A7	INCLUDE the effects of both normal and alternate system alignments, to the extent needed for determination of the frequency of each modeled event sequence and event sequence family. See Note SY-N-7	INCLUDE the effects of both normal and alternate system alignments. ENSURE normal and significant alternate system alignments are modeled. Asymmetrical modeling of trains is permitted if the trains, their support systems, and their underlying data have no significant differences in design and operation. See Note SY-N-7
SY-A8	In defining the system model boundary (see Requirement SY-A3), INCLUDE within the boundary the components required for system operation and the components providing the interfaces with support systems required for actuation and operation of the system components.	
SY-A9	DEVELOP detailed systems models, unless *(a)* sufficient system-level data are available to quantify the system failure probability; *(b)* system failure is dominated by operator actions or CCFs and omitting the model does not mask contributions to the results of support systems or other dependent failure modes; or *(c)* further detail is not needed to resolve risk-significant contributors and/or dependencies. For case (a), USE a single data value only for systems with no equipment or human action dependencies and if data exist that sufficiently represent the unreliability or unavailability of the system and capture plant- or design-specific factors that could influence unreliability and unavailability. JUSTIFY the use of limited (i.e., reduced or single data value) modeling. See Note SY-N-8	
SY-A10	For operating plants, CONFIRM that the level of detail in delineating the systems models is consistent with the as-built and as-operated plant sufficient to identify potential risk-significant contributors. See Note SY-N-6	
SY-A11	For PRAs performed during the pre-operational stage, CONFIRM that the level of detail in delineating the systems models is consistent with the design information and procedural guidance available sufficient to identify potential risk-significant contributors. See Note SY-N-4	

Table 4.3.5.1-2 Supporting Requirements for HLR-SY-A (Cont'd)

System logic models shall be developed that represent the various system alignments, success criteria, and mission times and include the failure modes associated with system maintenance, component actuation and functionality, and associated HFEs. (HLR-SY-A)

Index No. SY-A	Capability Category I	Capability Category II
SY-A12	DEFINE the boundaries of the components required for system operation in a way that is consistent with the component failure data. For example, a control circuit for a pump does not need to be included as a separate basic event (or events) in the system model if the pump failure data used in quantifying the system model include control circuit failures. MODEL as separate basic events, those subcomponents (e.g., a valve limit switch that is associated with a permissive signal for another component) that are shared by another component or affect another component, to address the dependent failure mechanism.	
SY-A13	For PRAs performed during the pre-operational stage, IDENTIFY uncertainties in the ability to match the modeled component boundaries with those associated with the applied generic data. See Note SY-N-4, SY-N-9	
SY-A14	If a system model is developed in which a single failure of a supercomponent (or module) is used to represent the collective impact of failures of several components, PERFORM the modularization process in a manner that avoids grouping events with different recovery potential, events that are required by other systems, or events that have probabilities that are dependent on the scenario. Examples of such events include the following: *(a)* hardware failures that are not recoverable versus actuation signals, which are recoverable; *(b)* HFEs that can have different probabilities dependent on the context of different event sequences; *(c)* events that are mutually exclusive of other events not in the module; *(d)* events that occur in other fault trees (especially CCFs); *(e)* structures, systems, and components (SSCs) used by other systems.	
SY-A15	INCLUDE the effect of variable success criteria (i.e., success criteria that change as a function of plant status or plant operating state) into the system modeling. Example causes of variable system success criteria include the following: *(a)* Different event scenarios or different plant operating states. Different success criteria are required for some systems to mitigate different event scenarios or the same event scenario occurring in a different plant operating state (e.g., the number of pumps required to operate in some systems is dependent upon the modeled initiating event or plant operating state); *(b)* Dependence on other components. Success criteria for some systems are also dependent on the success of another component in the system (e.g., operation of additional pumps in some cooling water systems is required if noncritical loads are not isolated); *(c)* Time dependence. Success criteria for some systems are time dependent (e.g., two pumps are required to provide the needed flow early following an event initiator, but only one is required for mitigation later following the event); *(d)* Sharing of a system between reactors or non-reactor sources of radioactive materials. Success criteria may be affected when two or more reactors in a multi-reactor plant are challenged by the same initiating event [e.g., loss of off-site power (LOOP)]; *(e)* Varying decay heat during a LPSD plant operating state. Changing decay heat with time, and after fuel reload, may affect system success criteria; *(f)* Variations in plant configurations (e.g., vessel level). Varying plant configurations affect the time to adverse plant conditions and the time available for system/component recovery.	

Table 4.3.5.1-2 Supporting Requirements for HLR-SY-A (Cont'd)

System logic models shall be developed that represent the various system alignments, success criteria, and mission times and include the failure modes associated with system maintenance, component actuation and functionality, and associated HFEs. (HLR-SY-A)

Index No. SY-A	Capability Category I	Capability Category II
SY-A16	INCLUDE in the system model those failures of the equipment and components that would affect system operability (as identified in the system success criteria), except when excluded using the criteria in Requirement SY-A20. This equipment includes both active components (e.g., pumps, valves, air compressors) and passive components (e.g., piping, heat exchangers, tanks) required for system operation. When identifying failure modes for the equipment and components in the model, ENSURE those listed in Requirement SY-A19 are reviewed for applicability.	
SY-A17	DO NOT INCLUDE in a system model component failures that would be beneficial to system operation, unless omission would distort the results. See Note SY-N-10	
SY-A18	INCLUDE those failures that can cause flow diversion pathways that result in failure to meet the system success criteria. See Note SY-N-11	
SY-A19	When identifying the failures in Requirement SY-A16, INCLUDE failure modes consistent with the applied success criteria, consistent with available data, design information, and model level of detail, except where excluded using the criteria in Requirement SY-A20. For example *(a)* active component fails to start; *(b)* active component fails to continue to run; *(c)* failure of a closed component to open; *(d)* failure of a closed component to remain closed; *(e)* failure of an open component to close; *(f)* failure of an open component to remain open; *(g)* active component spurious operation; *(h)* plugging of an active or passive component; *(i)* leakage of an active or passive component; *(j)* rupture of an active or passive component; *(k)* internal leakage of a component; *(l)* internal rupture of a component; *(m)* failure to provide signal/operate (e.g., instrumentation); *(n)* spurious signal/operation; *(o)* pre-initiator HFEs (see Requirement SY-A21); *(p)* other failures of a component to perform its required function.	
SY-A20	In meeting Requirements SY-A16, SY-A18, and SY-A19, DO NOT INCLUDE contributors to system unavailability and unreliability (i.e., components and specific failure modes) from the model only if one of the following screening criteria is met: *(a)* a component may be excluded from the system model if the total failure probability of the component failure modes resulting in the same effect on system operation is at least two orders of magnitude lower than the highest failure probability of the other components in the same system train that results in the same effect on system operation; *(b)* one or more failure modes for a component may be excluded from the systems model if the contribution to the total failure rate or probability is <1% of the total failure rate or probability for that component, when their effects on system operation are the same per the requirements of SCR-2 in Table 1.10-1, and only one system is impacted.	
SY-A21	In the system model, INCLUDE HFEs that cause the system or component to be unavailable when demanded. These events are referred to as pre-initiator human events. See also Human Reliability Analysis (HLR-HR-C)	

Table 4.3.5.1-2 Supporting Requirements for HLR-SY-A (Cont'd)

System logic models shall be developed that represent the various system alignments, success criteria, and mission times and include the failure modes associated with system maintenance, component actuation and functionality, and associated HFEs. (HLR-SY-A)

Index No. SY-A	Capability Category I	Capability Category II
SY-A22	For PRAs performed during the pre-operational stage, CONFIRM that the level of detail in delineating the pre-initiator and post-initiator HFEs is consistent with the level of detail of the design and procedure information sufficient to identify potential risk-significant contributors. See Note SY-N-4	
SY-A23	In the system model, INCLUDE HFEs that are expected during the operation of the system or component or that are accounted for in the final quantification of event sequences unless they are already included explicitly as events in the event sequence models. These HFEs are referred to as post-initiator human actions. See also Event Sequence Analysis (HLR-ES-A) and Human Reliability Analysis (HLR-HR-G) During LPSD and other non-full-power configurations, INCLUDE additional HFEs expected due to the different plant operating states.	
SY-A24	INCLUDE in either the system model or event sequence modeling those conditions that cause the system to isolate or trip, or those conditions that, once exceeded, cause the system to fail, or DEMONSTRATE that their exclusion does not impact the results. For example, conditions that isolate or trip a system include the following: *(a)* system-related parameters such as a high temperature within the system; *(b)* external parameters used to protect the system from other failures; *(c)* adverse environmental conditions (see Requirement SY-A29).	
SY-A25	In the systems model, INCLUDE out-of-service unavailability for components in the system model, unless excluded per Requirement SY-A20, consistent with the available design information and, in the case of operating plants, actual practices and history of the plant for removing equipment from service. Meet Requirement SY-A26, as applicable. See Note SY-N-12, SY-N-13	
SY-A26	For PRAs performed during the pre-operational stage, IDENTIFY assumptions in the system model made due to a lack of design details and operational experience required to satisfy SY-A25. See Note SY-N-4, SY-N-9	
SY-A27	INCLUDE events representing the simultaneous unavailability of redundant equipment when this is a result of planned activity (see Requirement DA-C18).	
SY-A28	IDENTIFY system conditions that cause a loss of desired system function (e.g., excessive heat loads, excessive electrical loads, excessive humidity, or changes in plant configuration or plant operating state).	
SY-A29	CONSIDER a conservative representation of system or component unavailability when the potential exists for rated or design capabilities to be exceeded. See Note SY-N-14	USE system or component unavailability when the potential exists for rated or design capabilities to be exceeded only if supported by one or more of the following: *(a)* test or operational data; *(b)* engineering analysis; *(c)* expert judgment (if using expert judgment, SATISFY the requirements of Section 4.2, Use of Expert Judgment). See Note SY-N-14
SY-A30	DEFINE system model nomenclature in a consistent manner to allow model manipulation and to represent the same designator when a component failure mode is used in multiple systems or trains.	

Table 4.3.5.1-2 Supporting Requirements for HLR-SY-A (Cont'd)

System logic models shall be developed that represent the various system alignments, success criteria, and mission times and include the failure modes associated with system maintenance, component actuation and functionality, and associated HFEs. (HLR-SY-A)

Index No. SY-A	Capability Category I	Capability Category II
SY-A31	DO NOT MODEL the repair of hardware faults, unless the probability of repair is justified through an analysis or examination of data. See Requirement DA-C20	
SY-A32	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the development of the Systems Analysis in a manner that supports the applicable requirements of HLR-ESQ-E.	
SY-A33	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Systems Analysis. See Note SY-N-4, SY-N-9	

Table 4.3.5.1-3 Supporting Requirements for HLR-SY-B

CCFs and both intersystem and intrasystem dependencies that could influence system performance shall be evaluated and shall be modeled as applicable, including evaluating functional, human, and phenomenological effects, as well as dependencies on plant operating states. (HLR-SY-B)

Index No. SY-B	Capability Category I	Capability Category II
SY-B1	MODEL intrasystem CCFs when supported by generic or plant- or design-specific data, or JUSTIFY that they are not a risk-significant contributor to system unreliability. See Note SY-N-15	MODEL intrasystem CCFs when supported by generic or plant- or design-specific data. See Note SY-N-15
SY-B2	Either *(a)* MODEL intersystem CCFs that are supported by applicable generic industry data; or *(b)* JUSTIFY that intersystem CCFs are not a risk-significant contributor to system unreliability. See Note SY-N-16	MODEL intersystem CCFs that are supported by applicable generic industry data and include plant-, technology-, or design-specific experience. See Note SY-N-16, SY-N-17
SY-B3	DEFINE CCF groups by using a logical, systematic process; JUSTIFY the basis for selecting CCF component groups by evaluating similarity in the following: *(a)* service conditions; *(b)* environment; *(c)* design or manufacturer; *(d)* maintenance. See Note SY-N-18	
SY-B4	INCLUDE CCFs into the system model consistent with the common cause model used for Data Analysis. See Requirement DA-D8	
SY-B5	INCLUDE dependency on support systems or interfacing systems in the modeling process.	
SY-B6	PERFORM engineering analyses to determine the need for support systems that are plant-specific and represent the variability in the conditions present during the event sequences for which the system is required to function.	
SY-B7	In support system modeling, USE conservative success criteria and timing.	In support system modeling, USE realistic success criteria and timing for risk-significant contributors.

Table 4.3.5.1-3 Supporting Requirements for HLR-SY-B (Cont'd)

CCFs and both intersystem and intrasystem dependencies that could influence system performance shall be evaluated and shall be modeled as applicable, including evaluating functional, human, and phenomenological effects, as well as dependencies on plant operating states. (HLR-SY-B)

Index No. SY-B	Capability Category I	Capability Category II
SY-B8	IDENTIFY spatial, environmental, and intrinsic hazards that may impact multiple systems or redundant components in the same system, and INCLUDE them in the system fault tree or the event sequence evaluation. During shutdown, temporary conditions may exist that may affect the ability of the modeled systems to respond to an initiating event. For each shutdown plant operating state, IDENTIFY these conditions, and INCLUDE the effect of the temporary condition on the operation of the system in the system model for the duration of the temporary condition. See Note SY-N-19, SY-N-20, SY-N-21	
SY-B9	When modeling a system, INCLUDE interfaces with the support systems required for successful operation of the system for a required mission time (see also Requirement SY-A8). See Note SY-N-22	
SY-B10	For PRAs performed during the pre-operational stage, CONFIRM that the level of detail in delineating the system interfaces is consistent with the design information available sufficient to identify potential risk-significant contributors (e.g., automatic actuation systems while at-power and manual actuation during shutdown operations). CONFIRM that assumptions regarding independence of systems or components where detailed design information is lacking are clearly documented, especially in cases where the systems or components are functionally redundant and unforeseen dependencies could lead to risk-significant impacts. See Note SY-N-4, SY-N-23	
SY-B11	JUSTIFY non-detailed modeling of systems that are required for initiation and actuation of a system (including software) in accordance with the criteria outlined in SY-A9.	MODEL those systems that are required for initiation and actuation of a system and software. For risk-significant systems, include the presence of the conditions needed for automatic actuation and the permissive and lock-out signals that are required to complete actuation logic to the extent supported by available design information. For non-risk-significant systems, meet the requirements of Capability Category I (CC-I).
SY-B12	MODEL the capability of the available inventories of air, power, and cooling to support the mission time. See Note SY-N-4, SY-N-24, SY-N-25	
SY-B13	DO NOT USE proceduralized recovery actions as the sole basis for eliminating a support system from the model (e.g., it is not acceptable to not model a system such as heating, ventilation, and air-conditioning or component cooling water on the basis that there are procedures for dealing with losses of these systems).	

Table 4.3.5.1-3 Supporting Requirements for HLR-SY-B (Cont'd)

CCFs and both intersystem and intrasystem dependencies that could influence system performance shall be evaluated and shall be modeled as applicable, including evaluating functional, human, and phenomenological effects, as well as dependencies on plant operating states. (HLR-SY-B)

Index No. SY-B	Capability Category I	Capability Category II
SY-B14	IDENTIFY SSCs that may be required to operate in conditions beyond their environmental qualifications. INCLUDE dependent failures of multiple SSCs that result from operation in these adverse conditions. Examples of degraded environments include the following: *(a)* reactor coolant boundary (RCB) or other radionuclide transport barrier breach in specific locations; *(b)* safety relief valve operability in harsh environments; *(c)* high-energy line breaks in different locations (e.g., steam-line breaks in areas that may impact modeled SSCs); *(d)* debris that could plug screens/filters (both internal and external to the plant); *(e)* heating of the water supply that could affect pump operability; *(f)* loss of net positive suction head for pumps; *(g)* steam binding or air binding of pumps; *(h)* loss of heating, ventilation, and air conditioning (HVAC) for digital I&C systems and other susceptible SSCs; *(i)* harsh environments induced by event sequence phenomena. See Note SY-N-3	
SY-B15	INCLUDE operator interface dependencies across systems, trains, or reactors (for multi-reactor sites), where applicable.	
SY-B16	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the development of the common cause, intersystem dependency, and intrasystem dependency Systems Analysis modeling in a manner that supports the applicable requirements of HLR-ESQ-E.	
SY-B17	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the CCF and dependency modeling. See Note SY-N-4, SY-N-9	

Table 4.3.5.1-4 Supporting Requirements for HLR-SY-C

The documentation of the Systems Analysis shall provide traceability of the work. (HLR-SY-C)

Index No. SY-C	Capability Category I	Capability Category II
SY-C1	DOCUMENT the process used in the Systems Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of Supporting Requirements (SRs) are satisfied: *(a)* system function and operation under normal and emergency operations; *(b)* system model boundary; *(c)* system schematic illustrating all equipment and components necessary for system operation; *(d)* information and calculations to support equipment operability considerations and assumptions; *(e)* actual operational history or history in similar systems indicating any past problems in the system operation; *(f)* system success criteria and relationship to event sequence models; *(g)* human actions necessary for operation of system; *(h)* reference to system-related test and maintenance procedures; *(i)* system dependencies and shared component interface; process used for systematic search for dependencies including dependency tables; *(j)* component spatial information, including spatial and environmental hazards that may impact multiple systems or redundant components in the same system; *(k)* assumptions or simplifications made in development of the system models; *(l)* the components and failure modes included in the model and justification for any exclusion of components and failure modes; *(m)* a description of the modularization process (if used); *(n)* records of resolution of logic loops developed during fault tree linking (if used); *(o)* results of the system model evaluations; *(p)* results of sensitivity studies (if used); *(q)* the sources of the above information (e.g., completed checklist from investigations, notes from discussions with plant or design personnel, as applicable); *(r)* basic events in the system fault trees so that they are traceable to modules and to cutsets; *(s)* the nomenclature used in the system models; *(t)* treatment of digital instrumentation and control systems; *(u)* treatment of systems that perform their safety functions using passive means, including documentation of the evaluation of uncertainties in system performance. See Note SY-N-26	
SY-C2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirements SY-A32 and SY-B16) associated with the Systems Analysis.	
SY-C3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Systems Analysis and, if applicable, assumption that the systems analyzed are free of design and construction errors or defects. See SY-A33 and SY-B17 See Note SY-N-4	

4.3.5.2 Peer Review Requirements for Systems Analysis

4.3.5.2.1 Purpose

This Section provides requirements for peer review of the Systems Analysis element of the PRA.

4.3.5.2.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the area of systems analysis. The team members assigned to review Systems Analysis shall overlap those that assigned to review Event Sequence Analysis to ensure consistency between the modeling for both elements. The team members assigned to review the Systems Analysis shall have experience specific to these areas and the capability of recognizing plant and design-specific features of the analyses.

4.3.5.2.3 Review of Systems Analysis to Confirm the Methodology

A review shall be performed on the Systems Analysis. The portion of system models selected for review typically includes a sample of the systems where failure contributes to risk-significant event sequences, including the following:

(a) different models representing different levels of detail;

(b) front-line system for each mitigating function (e.g., reactivity control, decay heat removal);

(c) each major type of support system (e.g., electrical power, cooling water, instrument air, and HVAC);

(d) complex system with variable success criteria (e.g., a cooling water system requiring different numbers of pumps for success dependent upon whether non-safety loads are isolated);

(e) systems performing functions relying on passive and inherent reactor features.

4.3.5.3 References for Systems Analysis

The following is a list of publications referenced in this Standard.

[SY-1] EPRI TR-113051, "An Analysis of loss of Decay Heat Removal Trends and Initialing Event Frequencies (1989–1988): Outage Risk Assessment and Management (ORAM) Technology," Electric Power Research Institute, 1999

[SY-2] EPRI TR-1003113, "An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989–2000): Outage Risk Assessment and Management (ORAM) Technology," Electric Power Research Institute, 2001

[SY-3] NUREG/CR-5485, "Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment," U.S. Nuclear Regulatory Commission, November 20, 1998

[SY-4] NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing," U.S. Nuclear Regulatory Commission, December 2007

NONMANDATORY APPENDIX SY: NOTES AND EXPLANATORY MATERIAL FOR SYSTEMS ANALYSIS

SY.1 NOTES ASSOCIATED WITH SYSTEMS ANALYSIS

Table SY-1 Notes Supporting Systems Analysis Requirements

Number	Notes
SY-N-1	The initial scope of systems in the PRA model will be established by the event sequence model and associated safety functions for the internal events hazard group. As seismic hazards are incorporated into the PRA model, there will be additional SSCs that are added to account for structures that support modeled internal event systems and to account for any new event sequences and safety functions that may be introduced by the seismic hazard. If there are other unscreened hazards added to the model, there may be additional SSCs that are added to the scope of the Systems Analysis. See SY-A1
SY-N-2	Examples of such information include (as applicable to the state of design) the following: *(a)* system piping and instrumentation drawings (P&IDs); *(b)* one-line diagrams; *(c)* I&C drawings; *(d)* spatial layout drawings; *(e)* system operating procedures; *(f)* success criteria calculations; *(g)* the preliminary, final, or updated safety analysis report (SAR); *(h)* Technical Specifications; *(i)* training information; *(j)* system descriptions and related design documents; *(k)* interviews with system designers, engineers, and/or operators; *(l)* actual system operating experience; and *(m)* abnormal operating procedures and emergency operating procedures. See SY-A2
SY-N-3	For LPSD states, look for outage-specific planning guides, temporary system alignments, and shutdown operating procedures. See SY-A2, SY-B14
SY-N-4	This SR is not applicable to operating plants. See SY-A4, SY-A6, SY-A11, SY-A13, SY-A22, SY-A26, SY-A33, SY-B10, SY-B12, SY-B17, SY-C3
SY-N-5	Additional systems investigations (e.g., tabletop review, physical/computerized walkdowns of the plant) would be necessary for systems and alignments not modeled in the Full-Power PRA. Systems that perform similar functions during LPSD and full-power conditions may not need additional investigations if included in the Full-Power PRA. See SY-A5, SY-A6
SY-N-6	This SR is likely not applicable to PRAs performed during the pre-operational stage. See SY-A5, SY-A10
SY-N-7	Many systems are realigned, tagged out, have their automatic functions disabled, etc., in the process of going into an outage. See SY-A7
SY-N-8	Examples of systems that have sometimes not been modeled in detail include the scram system, the power conversion system, instrument air, and the keep-fill systems. See SY-A9

Table SY-1 Notes Supporting Systems Analysis Requirements (Cont'd)

Number	Notes
SY-N-9	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See SY-A13, SY-A26, SY-A33, SY-B17
SY-N-10	An example of a beneficial failure: A failure of an instrument in such a fashion as to generate a required actuation signal. See SY-A17
SY-N-11	Unusual system alignments and infrequent operations, such as draindown, increase the potential for flow diversion pathways during shutdown conditions. See SY-A18
SY-N-12	Examples of out-of-service unavailability to be modeled are as follows (as applicable to each modeled plant operating state and depending on available design information): *(a)* unavailability caused by testing when a component or system train is reconfigured from its required event sequence mitigating position such that the component cannot function as required; *(b)* maintenance events at the train level when isolating the entire train for maintenance; *(c)* maintenance events at a subtrain level [i.e., between tagout boundaries, such as a functional equipment group (FEG)] when directed by procedures; *(d)* equipment intentionally taken out of service to change plant configurations or plant operating state such as LPSD to be modeled as failed or not credited; *(e)* equipment that will not function in certain plant operating states and is modeled as failed or not credited; *(f)* train outages during a work window for preventive/corrective maintenance; *(g)* an FEG removed from service for preventive/corrective maintenance; *(h)* a relief valve taken out of service. See SY-A25
SY-N-13	The capability to remove differing sets of SSCs for maintenance and testing is a unique characteristic of shutdown conditions. In some cases, because of the changes in maintenance configurations, additional plant operating states may need to be defined. See SY-A25
SY-N-14	During a shutdown plant operating state, for example, design capabilities for certain components may be challenged or exceeded. For CC-I, no credit for operability during these conditions should be applied. For Capability Category II (CC-II), credit for operability may be allowed if supported by plant abnormal procedures and engineering or vendor evaluations. See SY-A29
SY-N-15	An acceptable method is represented in NUREG/CR-5485 [SY-3]. See SY-B1
SY-N-16	Examples of intersystem CCFs include the following: bad fuel oil fails DGs in multiple systems, fish intrusion fails multiple systems drawing off same suction source, sump plugging fails multiple systems. See SY-B2
SY-N-17	Technology-specific testing experience includes testing experience collected from relevant nuclear and non-nuclear testing programs for the component or system of interest. An example includes experience collected from a separate effects test loop that will employ the same dependencies under approximately the same conditions as modeled in the PRA. See SY-B2

Table SY-1 Notes Supporting Systems Analysis Requirements (Cont'd)

Number	Notes
SY-N-18	Candidates for CCFs include, for example, *(a)* motor-operated valves; *(b)* pumps; *(c)* safety relief valves; *(d)* air-operated valves; *(e)* solenoid-operated valves; *(f)* check valves; *(g)* diesel generators; *(h)* batteries; *(i)* inverters and battery chargers; *(j)* circuit breakers; *(k)* digital I&C components and subsystems; *(l)* software components and subsystems. See SY-B3
SY-N-19	Examples of investigations include, but are not limited to, actives such as: tabletop review(s), physical/computerized walkdown(s) of the plant, and/or interview(s) with knowledgeable personnel. These investigation(s) may need to be supported by a review of plant documents, especially in the context of external hazard evaluations. This SR does not imply that each of these activities need to be performed to meet the SR. Further, if appropriate, one investigation can meet the intent of multiple investigation SRs if the investigation is scoped and structured appropriately. See SY-B8
SY-N-20	LPSD examples of these temporary conditions include the removal of flood, ventilation, or fire barriers, or the installation and removal of scaffolding or temporary shielding. See SY-B8
SY-N-21	The removal of a flood barrier as part of a maintenance activity may affect the ability of a modeled system or component to successfully perform its function in the event of an internal or external flooding event. These conditions are not typically included in an at-power PRA but are a consideration for modeling of shutdown plant operating states. See SY-B8
SY-N-22	Examples of support systems include the following: *(a)* actuation logic (often disabled during shutdown operations and activities) and digital I&C and associated software; *(b)* support systems required for control of components; *(c)* component motive power; *(d)* cooling of components; *(e)* any other identified support function (e.g., heat tracing) necessary to meet the success criteria and associated systems. See SY-B9
SY-N-23	NUREG-1860 [SY-4] provides an acceptable approach for searching for dependencies in new reactor designs. See SY-B10
SY-N-24	For PRAs performed during the pre-operational stage, model these inventories consistently with available design information. See SY-B12
SY-N-25	Inventories of air, cooling, and other services may be different in different plant operating states. See SY-B12
SY-N-26	An example of one method to satisfy this SR is a cross-reference identifying each requirement and where it is addressed in the documentation. This example of a documentation method facilitates PRA applications, upgrades, and peer reviews. See SY-C1

4.3.6 Human Reliability Analysis (HR)

This Section presents the technical requirements associated with Human Reliability Analysis.

4.3.6.1 Objectives and Technical Requirements for Human Reliability Analysis

The objectives of Human Reliability Analysis ensure that

(a) routine activities which can impact the availability of necessary equipment are identified for each plant operating state;

(b) unlikely errors are screened out;

(c) human failure events (HFEs) are defined for unscreened activities;

(d) probabilities of pre-initiator HFEs are assessed;

(e) a review of procedures, past events, procedural guidance, and training informs the identification of post-initiator operator responses;

(f) HFEs are defined that impact the event sequences progression;

(g) the probabilities of at- and post-initiator HFEs are assessed;

(h) plausible and feasible recovery actions are modeled; and

(i) the Human Reliability Analysis is documented to provide traceability of the work.

Table 4.3.6.1-1 High Level Requirements for Human Reliability Analysis

Designator	Requirement
HLR-HR-A	A systematic process shall be used to identify those specific routine activities that, if not completed correctly, may impact the availability of equipment necessary to perform system functions for each plant operating state modeled in the PRA.
HLR-HR-B	Screening out activities that need not be addressed explicitly in the model shall be based on an assessment of how reactor-specific and plant- or design-specific operational practices limit the likelihood of errors in such activities.
HLR-HR-C	For each activity that is not screened out, a HFE shall be defined for each applicable plant operating state to characterize the impact of the failure as an unavailability of a component, system, or function modeled in the PRA.
HLR-HR-D	The assessment of the probabilities of the pre-initiator HFEs shall be performed by using a systematic process that addresses the reactor-specific, plant- or design-specific, and activity-specific influences on human performance.
HLR-HR-E	A systematic review of the relevant available procedures, any past operational events, procedural guidance, and training shall be used to identify the set of post-initiator operator responses required for each of the event sequences.
HLR-HR-F	For each modeled plant operating state, HFEs shall be defined that represent the impact of not properly performing the required responses, in a manner consistent with the structure and level of detail of the event sequences.
HLR-HR-G	The assessment of the probabilities of the at-initiator and post-initiator HFEs shall be performed using a well-defined and self-consistent process that addresses the reactor-specific, plant- or design-specific, and scenario-specific influences on human performance and addresses potential dependencies between HFEs in the same event sequence.
HLR-HR-H	Recovery actions (at the cutset or scenario level) shall be modeled only if it has been demonstrated that the action is plausible and feasible for those scenarios to which they are applied. In this context, recovery is associated with operators performing actions to compensate for the failed automatic actions but does not include repair of the equipment.
HLR-HR-I	The documentation of the Human Reliability Analysis shall provide traceability of the work.

Table 4.3.6.1-2 Supporting Requirements for HLR-HR-A

A systematic process shall be used to identify those specific routine activities that, if not completed correctly, may impact the availability of equipment necessary to perform system functions for each plant operating state modeled in the PRA. (HLR-HR-A)

Index No. HR-A	Capability Category I	Capability Category II
HR-A1	For equipment and plant operating states modeled in the PRA, IDENTIFY through a review of procedures (if applicable), plant practices (if applicable) and industry operating experience those plant evolution activities and test, inspection, and maintenance activities that require realignment of equipment or a control system outside its normal operational or standby status. SELECT a representative plant evolution schedule for a list of test and maintenance activities to consider. Meet Requirement HR-A2, if applicable. See Note HR-N-1, HR-N-2	
HR-A2	For PRAs performed during the pre-operational stage, for equipment, and plant operating states modeled in the PRA, DEFINE and IMPLEMENT a process (e.g., interview with design engineers) for the identification of PRA equipment that requires realignment outside its normal operational or standby status. See Note HR-N-1, HR-N-3	
HR-A3	For equipment and plant operating states modeled in the PRA, IDENTIFY through a review of procedures (if applicable), plant practices (if applicable), and industry operating experience those calibration activities that, if performed incorrectly, can have an adverse impact on the automatic initiation of standby equipment or on the indications to operators of the need for manual actuation. SELECT a representative plant evolution schedule for a list of calibration activities to consider. Meet Requirement HR-A4, if applicable. See Note HR-N-2	
HR-A4	For PRAs performed during the pre-operational stage, for equipment and plant operating states modeled in the PRA, DEFINE and IMPLEMENT a process (e.g., interview with design engineers) for the identification of those calibration activities that if performed incorrectly can have an adverse impact on the automatic and/or manual initiation of standby safety equipment. See Note HR-N-3	
HR-A5	IDENTIFY work practices identified in Requirements HR-A1 and HR-A3 that involve an activity that simultaneously affects equipment in either different trains of a redundant system or diverse systems (e.g., use of common calibration equipment by the same crew on the same shift, a maintenance or test activity that requires realignment of an entire system). Meet Requirement HR-A6, as applicable. See Note HR-N-2	
HR-A6	For PRAs performed during the pre-operational stage, DEFINE and IMPLEMENT a process (e.g., interview with design engineers) used for the identification of the work practices identified in Requirements HR-A1 and HR-A3.	
HR-A7	For equipment and plant operating states modeled in the PRA, INCLUDE in the modeling of support system initiating fault trees (see Requirement IE-C11) the contribution of operator error leading to the initiating event, or PROVIDE the basis for the exclusion of such errors. See Note HR-N-4	
HR-A8	For PRAs performed during the pre-operational stage, ENSURE that the level of detail in delineating the operator actions is consistent with the design and procedure information available in a manner that is sufficient to identify potential risk-significant contributors. See Note HR-N-3	
HR-A9	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with pre-initiator Human Reliability Analysis in a manner that supports the applicable requirements of HLR-ESQ-E.	
HR-A10	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the pre-initiator Human Reliability Analysis. See Note HR-N-3, HR-N-5	

Table 4.3.6.1-3 Supporting Requirements for HLR-HR-B

Screening out activities that need not be addressed explicitly in the model shall be based on an assessment of how reactor-specific and plant- or design-specific operational practices limit the likelihood of errors in such activities. (HLR-HR-B)

Index No. HR-B	Capability Category I	Capability Category II
HR-B1	If screening is performed, SPECIFY criteria for screening out classes of activities from further analysis, per the requirements of SCR-3 in Table 1.10-1. ENSURE screening is applicable to each affected plant operating state. If screening is performed and if plant-specific procedures and finalized design information are not available, DEFINE the method used for screening out classes of activities from further analysis. See Note HR-N-6, HR-N-7	
HR-B2	JUSTIFY the applicability of screening when activities span multiple plant operating states when any related HFE would be detected by administrative controls before transitioning from one plant operating state to the next. See Note HR-N-8	
HR-B3	DO NOT SCREEN OUT activities that could simultaneously impact multiple trains of a redundant system or diverse systems (Requirement HR-A5). See Note HR-N-9	

Table 4.3.6.1-4 Supporting Requirements for HLR-HR-C

For each activity that is not screened out, a HFE shall be defined for each applicable plant operating state to characterize the impact of the failure as an unavailability of a component, system, or function modeled in the PRA. (HLR-HR-C)

Index No. HR-C	Capability Category I	Capability Category II
HR-C1	DEFINE a HFE that represents the impact of the human failures at the function, system, train, or component level for each activity that was not screened out per Requirement HR-B1. See Note HR-N-10	
HR-C2	IDENTIFY the average time to the detection of the failure considering administrative practices.	
HR-C3	For PRAs performed during the pre-operational stage, ENSURE that the level of detail in delineating the HFEs is consistent with the design and procedure information available in a manner that is sufficient to identify potential risk-significant contributors. See Note HR-N-3	
HR-C4	INCLUDE those modes of unavailability that, following completion of each retained activity, result from failure to restore the following: *(a)* equipment to the desired standby or operational status; *(b)* initiation signal or set point for equipment startup or realignment; *(c)* automatic realignment or power. See Note HR-N-11	INCLUDE those modes of unavailability that, following completion of each retained activity, result from failure to restore the following: *(a)* equipment to the desired standby or operational status; *(b)* initiation signal or set point for equipment startup or realignment; *(c)* automatic realignment or power. INCLUDE failure modes identified during the collection of plant- or design-specific or applicable generic operating experience that leave equipment unavailable for response in event sequences. See Note HR-N-11
HR-C5	INCLUDE the impact of miscalibration on operator performance and as a mode of failure of initiation of standby systems.	
HR-C6	For PRAs performed during the pre-operational stage, IDENTIFY assumptions in lieu of as-built and as-operated details that influence the delineation of HFEs. See Note HR-N-3, HR-N-5	

Table 4.3.6.1-5 Supporting Requirements for HLR-HR-D

The assessment of the probabilities of the pre-initiator HFEs shall be performed by using a systematic process that addresses the reactor-specific, plant- or design-specific, and activity-specific influences on human performance. (HLR-HR-D)

Index No. HR-D	Capability Category I	Capability Category II
HR-D1	SPECIFY the systematic process that will be used to determine the human error probabilities (HEPs). See Note HR-N-12	
HR-D2	USE conservative estimates in the quantification of the pre-initiator HEPs.	For risk-significant pre-initiator HFEs, USE detailed assessments in the quantification of pre-initiator HEP mean values. For non-risk-significant pre-initiator HFEs, ENSURE the requirement for Capability Category I (CC-I) is met.
HR-D3	For PRAs performed during the pre-operational stage, ENSURE that the level of detail in the assessments of pre-initiator HFE probabilities is consistent with the design and available procedure information in a manner that is sufficient to identify potential risk-significant contributors. See Note HR-N-3	
HR-D4	USE conservative estimates that take into account the factors related to the expected implementation of the human action. See Note HR-N-13	For each detailed HEP assessment, INCLUDE in the evaluation process the following plant- or design-specific relevant information when available: *(a)* the quality (e.g., format, logical structure, ease of use, clarity, comprehensiveness) of available written procedures (for performing tasks) and administrative controls that support independent review of written procedures (e.g., configuration control process, technical review process, training process, management emphasis on adherence to procedures); *(b)* the quality of the human-machine interface, including both the equipment configuration and instrumentation and control layout.
HR-D5	When addressing self-recovery or recovery from other crew members in estimating HEPs for specific HFEs, USE pre-initiator recovery factors consistent with selected methodology. If recovery of pre-initiator errors is credited, SPECIFY the maximum credit that can be given for multiple recovery opportunities.	
HR-D6	For operating plants, if recovery of pre-initiator errors is credited, USE the following information, if available, to assess the potential for recovery of pre-initiator errors: *(a)* post-maintenance or post-calibration tests required and proceduralized; *(b)* independent verification, using a hardcopy/electronic checklist, which verifies component status following maintenance/testing; *(c)* a separate verification of component status made at a later time, using a hardcopy/electronic checklist, by the original performer; *(d)* work shift or daily verifications of component status, using a hardcopy/electronic checklist. See Note HR-N-14	
HR-D7	EVALUATE the potential for dependencies of pre-initiator HFEs applicable to the modeled plant operating states (i.e., having some common elements in their causes, such as work performed by the same crew in the same time frame), and CALCULATE the joint probability of dependencies identified.	

Table 4.3.6.1-5 Supporting Requirements for HLR-HR-D (Cont'd)

The assessment of the probabilities of the pre-initiator HFEs shall be performed by using a systematic process that addresses the reactor-specific, plant- or design-specific, and activity-specific influences on human performance. (HLR-HR-D)

Index No. HR-D	Capability Category I	Capability Category II
HR-D8	CHARACTERIZE the uncertainty for the HEPs. See Note HR-N-15	For each risk-significant HFE, PROVIDE a probabilistic representation of the uncertainty of the calculated HEPs. For the HFEs that are not risk-significant, CHARACTERIZE the uncertainty. See Note HR-N-15
HR-D9	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the assessment of the probabilities of the pre-initiator HFEs in a manner that supports the applicable requirements of HLR-ESQ-E.	
HR-D10	For PRAs performed during the pre-operational stage, IDENTIFY assumptions in lieu of as-built and as-operated details that influence the assessment of the probabilities of the pre-initiator HFEs. See Note HR-N-3, HR-N-5	

Table 4.3.6.1-6 Supporting Requirements for HLR-HR-E

A systematic review of the relevant available procedures, any past operational events, procedural guidance, and training shall be used to identify the set of post-initiator operator responses required for each of the event sequences. (HLR-HR-E)

Index No. HR-E	Capability Category I	Capability Category II
HR-E1	For operating plants, when identifying the key human response actions for each of the event sequences, REVIEW *(a)* the plant-specific emergency operating procedures (EOPs) and other relevant procedures (e.g., abnormal operating procedures or AOPs, annunciator response procedures) in the context of the scenarios; *(b)* system operation such that the system(s) functions and the human interfaces with the system are understood; *(c)* past operational events (both for the specific plant and industry) to assist the analyst in identifying the kinds of activities that have resulted in real-world HFEs. See Note HR-N-14, HR-N-16	
HR-E2	If plant-specific procedures and finalized design information are not available, when identifying the key human response actions, REVIEW the following: *(a)* the available or planned EOPs and other relevant procedures (e.g., AOPs, annunciator response procedures, or planned operational approach), as applicable, in the context of the scenarios; *(b)* system operation such that the system(s) functions and the human interfaces with the system are understood; *(c)* assumptions made in lieu of actual procedures with individuals responsible for plant operations or training, or engineering if plant operations or training individuals are not available, to confirm that interpretation is consistent with expected operational and training practices. See Note HR-N-3, HR-N-16	
HR-E3	For PRAs performed during the pre-operational stage, IDENTIFY, REVIEW, and USE (as appropriate) relevant, available, and applicable experience from other facilities (potentially including nonnuclear facilities, fuel cycle facilities, and nonpower reactors) to support the identification of post-initiator HFEs. See Note HR-N-3	

Table 4.3.6.1-6 Supporting Requirements for HLR-HR-E (Cont'd)

A systematic review of the relevant available procedures, any past operational events, procedural guidance, and training shall be used to identify the set of post-initiator operator responses required for each of the event sequences. (HLR-HR-E)

Index No. HR-E	Capability Category I	Capability Category II
HR-E4	IDENTIFY *(a)* those actions required to initiate (for those systems not automatically initiated), operate, control, isolate, or terminate those systems and components used in preventing or mitigating an event sequence as defined by the success criteria (e.g., operator initiates decay heat removal system); *(b)* those actions performed by the control room staff either in response to procedural direction or as skill-of-the-craft to diagnose and then recover a failed function, system, or component that is used in the performance of a response action as identified in Requirement HR-E1.	
HR-E5	For operating plants, REVIEW the interpretation of the procedures with plant operations or training personnel to confirm that interpretation is consistent with plant operating and training practices. See Note HR-N-14	For operating plants, USE talk-throughs (i.e., review in detail) with plant operations and training personnel the procedures and sequence of events to confirm that interpretation of the procedures is consistent with plant observations and training procedures. See Note HR-N-14
HR-E6	For PRAs performed during the pre-operational stage, REVIEW available procedures, procedural guidance, or assumptions made in lieu of procedures with individuals responsible for plant operations or training, to confirm that interpretation is consistent with expected operational and training practices. See Note HR-N-3	For PRAs performed during the pre-operational stage, USE talk-throughs (i.e., review in detail) with knowledgeable personnel associated with the development of plant operations or plant personnel training programs, to confirm that interpretation is consistent with the expected operation and training practices. See Note HR-N-3
HR-E7	REVIEW the interpretation of the human response with personnel knowledgeable about operations or personnel training to check that interpretation is consistent with expected human response.	USE simulation observations or talk-throughs with knowledgeable personnel to confirm the human response actions for scenarios modeled.
HR-E8	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the identification of the set of post-initiator operator responses in a manner that supports the applicable requirements of HLR-ESQ-E.	
HR-E9	For PRAs performed during the pre-operational stage, IDENTIFY assumptions in lieu of as-built and as-operated details that influence the identification of the set of post-initiator operator responses. See Note HR-N-3, HR-N-5	

Table 4.3.6.1-7 Supporting Requirements for HLR-HR-F

For each modeled plant operating state, HFEs shall be defined that represent the impact of not properly performing the required responses in a manner consistent with the structure and level of detail of the event sequences. (HLR-HR-F)

Index No. HR-F	Capability Category I	Capability Category II
HR-F1	DEFINE HFEs that represent the impact of the human failures at the function, system, train, or component level as appropriate. Failures to correctly perform several responses may be grouped into one HFE if the impact of the failures is similar or can be conservatively bounded. See Note HR-N-17	
HR-F2	For PRAs performed during the pre-operational stage, ENSURE that the level of detail in defining the HFEs is consistent with the design and procedure information available in a manner that is sufficient to identify potential risk-significant contributors. See Note HR-N-3	
HR-F3	GROUP failures to correctly perform the same response for different plant operating states across all plant operating states ONLY if the boundary conditions for the limiting plant operating state is used. See Note HR-N-18	GROUP failures to correctly perform the same response for different plant operating states ONLY for those plant operating states where the HFE boundary conditions are the same, or if the HFEs with the same response are not risk-significant and the boundary conditions of the limiting plant operating state for the group are used to represent the group. See Note HR-N-18
HR-F4	SPECIFY for the defined HFEs *(a)* plant operating state and event sequence-specific timing of cues and success criteria for the HFE (e.g., time window); *(b)* event sequence-specific procedural guidance (e.g., abnormal operating procedures or AOPs, emergency operating procedures, or EOPs); *(c)* the availability of cues and other indications for detection and evaluation errors; *(d)* the complexity of implementing the actions and the operator's training and experience about the implementation. (Task analysis is not required.)	SPECIFY for the defined HFEs *(a)* plant operating state and event sequence-specific timing of cues and success criteria for the HFE (e.g., time window); *(b)* event sequence-specific procedural guidance (e.g., AOPs, EOPs); *(c)* the specific high-level tasks (e.g., train level) required to complete the HFE; *(d)* the availability of cues and information needed to initiate and to complete each high-level task of the HFE; *(e)* the complexity and both the operator's training and experience about implementing each high-level task.
HR-F5	For PRAs performed during the pre-operational stage, IDENTIFY assumptions in lieu of as-built and as-operated details that influence the definition of HFEs that represent the impact of not properly performing required responses. See Note HR-N-3, HR-N-5	

Table 4.3.6.1-8 Supporting Requirements for HLR-HR-G

The assessment of the probabilities of the at-initiator and post-initiator HFEs shall be performed using a well-defined and self-consistent process that addresses the reactor-specific, plant- or design-specific, and scenario-specific influences on human performance and addresses potential dependencies between HFEs in the same event sequence. (HLR-HR-G)

Index No. HR-G	Capability Category I	Capability Category II
HR-G1	USE conservative estimates for the HEPs of the HFEs in the event sequences that survive initial quantification.	PERFORM detailed analyses for estimation of HEPs for risk-significant HFEs. For the HEPs of HFEs that are not risk-significant, ENSURE the requirement for CC-I is met.
HR-G2	For PRAs performed during the pre-operational stage, ENSURE that the level of detail in the assessments of at-initiator and post-initiator HFE probabilities is consistent with the design and procedure information available in a manner that is sufficient to identify potential risk-significant contributors. See Note HR-N-3	
HR-G3	USE an approach to estimation of HEPs that addresses failure in cognition as well as failure to execute.	
HR-G4	When estimating HEPs, ADDRESS the following: *(a)* the complexity of detection, diagnosis, decision-making, and executing the required response; *(b)* the time available and time required to complete the response; *(c)* some measure of scenario-induced stress. See Note HR-N-19, HR-N-20	When estimating HEPs, EVALUATE the impact of the following plant- or design-specific and scenario-specific performance-shaping factors: *(a)* quality, including type (classroom or simulator) and frequency, of the operator training or experience, including just-in-time training prior to complex plant evolutions; *(b)* quality of the written procedures and administrative controls for the applicable plant operating states; *(c)* availability of instrumentation needed to take corrective actions; *(d)* degree of clarity of cues/indications; *(e)* human-machine interface; *(f)* time available and time required to complete the response; *(g)* complexity of detection, diagnosis, decision-making, and executing the required response; *(h)* environment (e.g., harsh conditions such as reduced lighting, extreme heat or cold, radiation) under which the operator is working; *(i)* accessibility of the equipment requiring manipulation; *(j)* physical demands of a task, including resistance to physical movement of equipment requiring manipulation; *(k)* staffing availability; *(l)* necessity, adequacy, and availability of special tools, parts, clothing, etc.; *(m)* mental fatigue associated with long-lasting event scenarios; *(n)* time pressure (sense of urgency) and stress (e.g., concern for family, personal safety) as perceived by operators; *(o)* for plant designs with remote or autonomous operations, operability of network connection, etc. See Note HR-N-19

Table 4.3.6.1-8 Supporting Requirements for HLR-HR-G (Cont'd)

The assessment of the probabilities of the at-initiator and post-initiator HFEs shall be performed using a well-defined and self-consistent process that addresses the reactor-specific, plant- or design-specific, and scenario-specific influences on human performance and addresses potential dependencies between HFEs in the same event sequence. (HLR-HR-G)

Index No. HR-G	Capability Category I	Capability Category II
HR-G5	ESTIMATE HEPs assuming all plant indications needed to alert the operators to take corrective actions within the context of the relevant event sequence are functioning properly.	When estimating HEPs, EVALUATE the availability of indications of plant conditions needed to alert the operators to take corrective actions within the context of the relevant event sequence, consistent with Requirement HR-A5.
HR-G6	For the time available to complete actions in each applicable plant operating state, USE applicable generic studies (e.g., thermal-hydraulic analysis for similar plants).	For the time available to complete actions in each applicable plant operating state, USE design- or plant-specific evaluations, appropriate realistic thermal-hydraulic analyses from similar plants (i.e., designs or plants of similar design and operation), or simulations from similar designs or plants.
HR-G7	For the time available to complete actions in each applicable plant operating state, SPECIFY the point in time at which operators are expected to receive relevant indications.	
HR-G8	When needed for the calculation of an HEP, ESTIMATE the time required to complete actions. See Note HR-N-20	For risk-significant HFEs, ESTIMATE the time required to complete the action based on action-time measurements in either investigation(s) or talk-throughs of either procedures or simulation observations. For non-risk-significant HFEs, ENSURE the requirements for CC-I are met. See Note HR-N-21
HR-G9	For operating plants, ENSURE the consistency of the at-initiator and post-initiator HEP quantifications. REVIEW the HFEs and their final HEPs relative to each other to ensure their reasonableness given the scenario context, plant history, procedures, operational practices, and experience. See Note HR-N-14	
HR-G10	For PRAs performed during the pre-operational stage, ENSURE the consistency of the at-initiator and post-initiator HEP quantifications. CONFIRM the consistency and reasonableness of the HEPs in light of similar plant(s) and scenario context. See Note HR-N-3	
HR-G11	DEFINE a minimum value for the joint probability of multiple human errors occurring in a given cutset or event sequence and JUSTIFY the minimum value to be used for the joint probability of multiple human errors occurring for a given cutset or event sequence. See Note HR-N-22	

Table 4.3.6.1-8 Supporting Requirements for HLR-HR-G (Cont'd)

The assessment of the probabilities of the at-initiator and post-initiator HFEs shall be performed using a well-defined and self-consistent process that addresses the reactor-specific, plant- or design-specific, and scenario-specific influences on human performance and addresses potential dependencies between HFEs in the same event sequence. (HLR-HR-G)

Index No. HR-G	Capability Category I	Capability Category II
HR-G12	For multiple human actions in the same event sequence or cutset (including at-initiator and post-initiator HFEs and recovery actions as per Requirement HR-H5), ASSESS the degree of dependence effect for each plant operating state and CALCULATE a joint human error probability. INCLUDE the influence of success or failure in preceding HFEs and system performance on the HFE being analyzed, including the following: *(a)* the outcome of the preceding HFEs that directly affect the performance of the HFE being analyzed [e.g., taking a longer time to implement, requiring the performance of more tasks, or a common cause (e.g., using the same procedure) that failed one or more of the preceding HFEs would affect the failure probability of the HFE being analyzed]; *(b)* the sharing of the same limited resources for the multiple HFEs (e.g., people, equipment, and water supply), and their performance has time overlap; *(c)* the sharing of the same cognitive behavior for the multiple HFEs, such as having a biased mindset about the condition or lack of questioning attitude [e.g., the stop, think, act, and review (STAR)]. See Note HR-N-23	
HR-G13	For each plant operating state and for multiple human actions in the same event sequence or cutset, if the joint human error probability calculated per HR-G12 is below the minimum value from HR-G11, USE the minimum value or PROVIDE the technical justification for the use of the lower joint probability based on an applicable evaluation of each cutset or event sequence with that combination.	
HR-G14	CALCULATE a point estimate HEP for each HFE. CHARACTERIZE the uncertainty for the calculated HEPs. See Note HR-N-15	CALCULATE a mean HEP for each risk-significant HFE. PROVIDE a probabilistic representation of the uncertainty of the calculated HEPs. For the HFEs that are not risk-significant, ENSURE the requirement for CC-I is met. See Note HR-N-15
HR-G15	IDENTIFY in the sources of uncertainty those factors impacting the level of dependence between the pre-initiator and post-initiator HFEs.	
HR-G16	For PRAs performed during the pre-operational stage, IDENTIFY assumptions in lieu of as-built and as-operated details that influence the assessment of the probabilities of the post-initiator HFEs. See Note HR-N-3, HR-N-5	

Table 4.3.6.1-9 Supporting Requirements for HLR-HR-H

Recovery actions (at the cutset or scenario level) shall be modeled only if it has been demonstrated that the action is plausible and feasible for those scenarios to which they are applied. In this context, recovery is associated with operators performing actions to compensate for the failed automatic actions but does not include repair of the equipment. (HLR-HR-H)

Index No. HR-H	Capability Category I	Capability Category II
HR-H1	IDENTIFY operator recovery actions that can restore the functions, systems, or components as needed to provide a more realistic evaluation of risk-significant event sequences. See Note HR-N-24	
HR-H2	DEFINE operator recovery actions only if, on a plant- or design-specific basis, they are assessed to be feasible, including the following: *(a)* a procedure or procedural guidance is available, and operator training has included the action as part of the crew's training, or justification for the omission for one or both is provided; *(b)* "cues" (e.g., alarms) that alert the operator to the recovery action provided procedure, training, or skill of the craft exist; *(c)* attention is given to the relevant performance-shaping factors provided in Requirement HR-G4; *(d)* there is sufficient manpower and time available to perform the action; *(e)* any environmental effects (e.g., high temperatures, loss of visibility) associated with the scenario do not preclude the recovery actions. Meet the requirement in HR-H3, if applicable.	
HR-H3	For PRAs performed during the pre-operational stage, JUSTIFY assumptions made to satisfy HR-H2 that is used to support the modeling of recovery actions; otherwise, DO NOT INCLUDE recovery actions in the PRA model. See Note HR-N-3	
HR-H4	ESTIMATE the HEPs for the operator recovery actions in a manner consistent with the CC-I requirements of HR-G1, HR-G2, HR-G3, HR-G4, HR-G5, HR-G6, HR-G7, HR-G8, HR-G9, and HR-G14.	ESTIMATE the HEPs for the operator recovery actions in a manner consistent with the Capability Category II (CC-II) requirements of HR-G1, HR-G2, HR-G3, HR-G4, HR-G5, HR-G6, HR-G7, HR-G8, HR-G9, and HR-G14.
HR-H5	INCLUDE any dependency between the HFE for operator recovery and any other HFEs in the sequence, scenario, or cutset to which the recovery is applied, including dependence with an HFE causing the initiating event. See HR-G11, HR-G12, HR-G13	
HR-H6	For PRAs performed during the pre-operational stage, IDENTIFY assumptions in lieu of as-built and as-operated details that influence the assessment of recovery actions. See Note HR-N-3, HR-N-5	

Table 4.3.6.1-10 Supporting Requirements for HLR-HR-I (Cont'd)

The documentation of the Human Reliability Analysis shall provide traceability of the work. (HLR-HR-I)

Index No. HR-I	Capability Category I	Capability Category II
HR-I1	DOCUMENT the process used in the Human Reliability Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of Supporting Requirements (SRs) are satisfied: *(a)* Human Reliability Analysis methodology and process used to identify pre-initiator and post-initiator HFEs, including how that method was adapted, where necessary, to account for the caveats at Requirement HR-D1; *(b)* qualitative screening rules and results of screening; *(c)* factors used in the quantification of the human action, how they were derived, their bases, and how they were incorporated into the quantification process; *(d)* quantification of HFEs, including the following: *(1)* screening values and their bases, *(2)* detailed HFE analyses with uncertainties and their bases, *(3)* the method and treatment of dependencies for pre-initiator and post-initiator actions, including justification of joint probabilities, given the caveats at Requirement HR-D1, *(4)* tables of pre-initiator, at-initiator, and post-initiator human actions evaluated by model, plant operating state, system, initiating event, and function, *(5)* HFEs for recovery actions and their dependency with other HFEs.	
HR-I2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirements HR-A9, HR-D9, HR-E8, and HR-G15) associated with the Human Reliability Analysis.	
HR-I3	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Human Reliability Analysis. See HR-A10, HR-C6, HR-D10, HR-E9, HR-F5, HR-G16, and HR-H6 See Note HR-N-3	

4.3.6.2 Peer Review Requirements for Human Reliability Analysis

4.3.6.2.1 Purpose

This Section provides requirements for peer review of the Human Reliability Analysis element of the PRA.

4.3.6.2.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the area of human reliability analysis. The team members assigned to review Human Reliability Analysis shall overlap that assigned to review Systems Analysis and Event Sequence Analysis to ensure consistency between the modeling for all elements. The team members assigned to review the Human Reliability Analysis shall have experience specific to these areas and the capability of recognizing plant and design-specific features of the analyses.

4.3.6.2.3 Review of Human Reliability Analysis to Confirm the Methodology

A review shall be performed on the Human Reliability Analysis.

The portion of the Human Reliability Analysis selected for review typically includes a sample of the HFEs whose failure contributes to risk-significant event sequences, including the following:

(a) the selection and implementation of any conservative HEPs used in the PRA;

(b) post-initiator HFEs and associated HEPs;

(c) pre-initiator HFEs and associated HEPs for both instrumentation miscalibration and failure of equipment;

(d) HEPs for the same function but under the influence of different performance shaping factors (PSFs);

(e) HEPs for dependent human actions, including dependencies of multiple HEPs in the same sequence;

(f) HEPs less than 1E-4;

(g) HFEs and associated HEPs involving remote actions in harsh environments;

(h) the selection and identification of the HFEs associated with the HEPs for the above review topics.

4.3.6.3 References for Human Reliability Analysis

The following is a list of publications referenced in this Standard.

[HR-1] EPRI TP-101711, D. T. WAKEFIELD, G. W. PARRY, G. W. HANNAMAN, and A. J. SPURGIN, “SHARP—A Revised Systematic Human Action Reliability Procedure,” Electric Power Research Institute, 1992

[HR-2] NUREG/CR-1278, SAND80-0200, A. D. SWAIN and H. E. GUTTMANN, “Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications,” U.S. Nuclear Regulatory Commission/Sandia National Laboratories, August 1983

[HR-3] NUREG/CR-4772, A. D. SWAIN, “Accident Sequence Evaluation Program Human Reliability Analysis Procedure,” U.S. Nuclear Regulatory Commission, February 1987

[HR-4] NUREG-0700, “Human-System Interface Design Review Guidelines,” U.S. Nuclear Regulatory Commission, July 2020

[HR-5] EPRI TR-1003113, “An Analysis of Loss of Decay Heat Removal Trends and Initiating Event Frequencies (1989–2000): Outage Risk Assessment and Management (ORAM) Technology,” Electric Power Research Institute, 2001

[HR-6] EPRI TR-1021081, “Establishing Minimum Acceptable Values for Probabilities of Human Failure Events: Practical Guidance for PRA,” Electric Power Research Institute, 2010

[HR-7] EPRI TR-3002003150, “A Process for HRA Dependency and Considerations on Use of Minimum Values for Joint Human Error Analysis,” Electric Power Research Institute, 2016

[HR-8] NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA), U.S. Nuclear Regulatory Commission, April 2005

NONMANDATORY APPENDIX HR: NOTES AND EXPLANATORY MATERIAL FOR HUMAN RELIABILITY ANALYSIS

HR.1 NOTES ASSOCIATED WITH HUMAN RELIABILITY ANALYSIS

Table HR-1 Notes Supporting Human Reliability Analysis Requirements

Number	Notes
HR-N-1	Plant evolution activities are those that are routinely performed during each plant operating state of the evolution. These include activities to align or block systems from service and to affect plant operating state transitions. See HR-A1, HR-A2
HR-N-2	Plant-specific procedures and practices may not be available to pre-operational plants, and thus, these procedures are supplemented by the information in the linked SR. See HR-A1, HR-A3, HR-A5
HR-N-3	This SR is not applicable to operating plants. See HR-A2, HR-A4, HR-A8, HR-A10, HR-C3, HR-C6, HR-D3, HR-D10, HR-E2, HR-E3, HR-E6, HR-E9, HR-F2, HR-F5, HR-G2, HR-G10, HR-G16, HR-H3, HR-H6, HR-I3
HR-N-4	A useful reference for identifying initiating events during shutdown is EPRI TR-1003113 [HR-5]. Although this reference was developed for operating LWR plants, it may provide useful guidance for evaluating non-LWR designs. See HR-A7
HR-N-5	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See HR-A10, HR-C6, HR-D10, HR-E9, HR-F5, HR-G16, HR-H6
HR-N-6	Example: Screen out maintenance and test activities from further analysis if the plant practices are generally structured to include independent verification of restoration of equipment to standby or operational status on completion of the activity. See HR-B1
HR-N-7	Screening can only be done on a plant operating state-by-plant operating state basis; i.e., the screening criteria are to be met for each particular plant operating state, for the activity to be screened. See HR-B1
HR-N-8	HFEs associated with pre-initiator activities (see the SRs of HLR-HR-C) may occur in one plant operating state and remain undetected in subsequent plant operating states. The impacts of such activities can be omitted from subsequent plant operating states provided they would be detected before entry into the later plant operating state. The impacts of the HFEs, of course, need not be applied in earlier plant operating states. See HR-B2
HR-N-9	During LPSD evolutions, only one train may be protected during maintenance on multiple systems of the other train(s). See HR-B3
HR-N-10	The defined pre-initiator HFEs apply to the associated plant operating state in which each is performed. Pre-initiator HFEs performed in one plant operating state may influence the plant response to events initiated in later plant operating states. Post-maintenance restoration tests may be delayed during plant evolutions. The time to detection of the pre-initiator HFE can determine the extent of impact in terms of which subsequent plant operating states may be affected. See HR-C1
HR-N-11	Miscalibration can be especially troublesome with respect to operator performance if only one train of equipment is available (e.g., it can lead to a so-called error of commission in stopping running equipment). See HR-C4

Table HR-1 Notes Supporting Human Reliability Analysis Requirements (Cont'd)

Number	Notes
HR-N-12	While standard human reliability analysis methods may be appropriate for pre-initiator events during LPSD evolutions, some adaptation of the method inputs may be required for the conditions existing during LPSD evolutions. See HR-D1
HR-N-13	Examples of factors related to the expected implementation of the human action can include the following: *(a)* the operating philosophy; *(b)* pre-operational stage human response assessments; *(c)* the quality of existing written procedures; *(d)* the quality of administrative controls; *(e)* human-machine interfaces. See HR-D4
HR-N-14	This SR is likely not applicable to PRAs performed during the pre-operational stage as they are assumed not to have access to adequate plant history, procedures, operational practices, and experience. See HR-D6, HR-E1, HR-E5, HR-G9
HR-N-15	This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding. See HR-D8, HR-G14
HR-N-16	Reference [HR-5] provides a list of numerous events during LPSD evolutions. As procedure reviews are specialized to LPSD evolution conditions, *(a)* reviews can only be done for the plant operating states applicable to each procedure as conditions, cues to operators, and procedural guidance can vary widely among plant operating states; *(b)* procedures applicable during LPSD evolutions have much less practical verification than at-power evolution procedures; be sure to search for traps (places where the procedure can dead-end, loop, or skip needed steps), and discuss them with operators and maintenance personnel consistent with Requirement HR-E5. See HR-E1, HR-E2
HR-N-17	The grouping of responses here refers to the combination of HFEs with the same impacts on the plant at the function, system, train, or component level. Consideration of grouping such actions across plant operating states is instead the subject of Requirement HR-F3. See HR-F1
HR-N-18	Here, HFE "boundary conditions" refers to both the impact of the HFEs as required by Requirement HR-F1 and the conditions needed to complete the HFE definitions as required in Requirement HR-F4. See HR-F3
HR-N-19	See SRs of HLR-HR-C for scenario-specific factors that should be considered when analyzing post-initiator HFEs. See HR-G4
HR-N-20	The Accident Sequence Evaluation Program (ASEP) approach [HR-3] is an acceptable approach. See HR-G4, HR-G8
HR-N-21	Examples of investigations include, but are not limited to, actives such as: tabletop review(s), physical/computerized walkdown(s) of the plant, and/or interview(s) with knowledgeable personnel. These investigation(s) may need to be supported by a review of plant documents, especially in the context of external hazard evaluations. This SR does not imply that each of these activities need to be performed to meet the SR. Further, if appropriate, one investigation can meet the intent of multiple investigation SRs if the investigation is scoped and structured appropriately. See HR-G8

Table HR-1 Notes Supporting Human Reliability Analysis Requirements (Cont'd)

Number	Notes
HR-N-22	When considering multiple human actions in the same event sequence or cutset, a minimum joint human error probability should be identified and justified. The minimum joint HEP should be based on the context of the event sequence, where the context is defined by the states of the performance-shaping factors (e.g., those required to be evaluated in HR-G4). One approach for establishing minimum HEP values is provided in EPRI 1021081 [HR-6] or the updated version, EPRI 3002003150 [HR-7]. NUREG-1792 [HR-8] also provides a discussion of the minimum joint HEP. See HR-G11
HR-N-23	The state-of-the-art in human reliability analysis is such that the assessment of dependency is largely based on the analyst's judgment. While it should be expected that there will be a progressively more detailed treatment of dependency in going from CC-I to CC-II, the distinction is not made at the level of this SR. Instead, it is expected to follow from the increase in the level of detail in the analysis of HFEs in going from CC-I to CC-II. Based on the identified dependencies and the larger number of manual actions during low-power-and-shutdown evolutions, justification of the joint probability of multiple actions is especially important. Note that dependency can vary between plant operating states and can occur across plant operating states for actions whose latent impacts (e.g., due to pre-initiator HFEs) are applicable to subsequent plant operating states. See HR-G12
HR-N-24	Recovery actions are actions taken in addition to those normally identified in the review of emergency, abnormal, and system operating procedures, which would normally be addressed in the SRs of HLR-HR-F and HLR-HR-G. They are included to allow credit for recovery from failures in cutsets or scenarios when failure to take credit would distort the insights from the risk analysis. The potential for recovery (e.g., manually opening a valve that had failed to open automatically) may well differ from scenario to scenario or cutset to cutset. In this context, recovery is associated with workarounds but does not include repair, which is addressed in Requirements SY-A31 and DA-C20. Restoration may be limited by pre-initiator activities leaving equipment in unavailable states. See HR-H1

4.3.7 Data Analysis (DA)

This Section presents the technical requirements associated with Data Analysis.

4.3.7.1 Objectives and Technical Requirements for Data Analysis

The objectives of Data Analysis ensure that

(a) parameter boundaries are defined;
(b) components are appropriately grouped;
(c) parameter data are consistent with parameter definitions;
(d) relevant generic industry and plant-specific evidence are represented in the parameter estimation, including addressing uncertainties; and
(e) the Data Analysis is documented to provide traceability of the work.

Table 4.3.7.1-1 High Level Requirements for Data Analysis

Designator	Requirement
HLR-DA-A	Each parameter shall be clearly defined in terms of the logic model, the basic event boundary, and the model used to evaluate event probability.
HLR-DA-B	Grouping components into a homogeneous population for parameter estimation shall consider the design, environmental, and service conditions of the components in the as-built and as-operated plant.
HLR-DA-C	Generic parameter estimates shall be chosen, and plant-specific data shall be collected consistent with the parameter definitions of the SRs of HLR-DA-A and the grouping rationale of the SRs of HLR-DA-B.
HLR-DA-D	The parameter estimates shall be based on relevant generic industry and technology- and plant-specific evidence. Where feasible, generic and technology- and plant-specific evidence shall be integrated using acceptable methods to obtain technology- and plant-specific parameter estimates. Each parameter estimate shall be accompanied by a characterization of the uncertainty.
HLR-DA-E	The documentation of the Data Analysis shall provide traceability of the work.

Table 4.3.7.1-2 Supporting Requirements for HLR-DA-A

Each parameter shall be clearly defined in terms of the logic model, the basic event boundary, and the model used to evaluate event probability. (HLR-DA-A)

Index No. DA-A	Capability Category I	Capability Category II
DA-A1	IDENTIFY from the Systems Analysis the basic events for which probabilities are required. See Note DA-N-1	
DA-A2	DEFINE structure, system, and component (SSC) boundaries; failure modes; and success criteria consistent with corresponding basic event definitions in Systems Analysis (Requirements SY-A7, SY-A9, SY-A12, SY-A14, SY-A15, SY-A16, SY-A17, and SY-B4) for failure rates and CCF parameters as applicable for each modeled plant operating state, and DEFINE boundaries of unavailability events consistent with corresponding definitions in Systems Analysis (Requirement SY-A25).	
DA-A3	USE an appropriate probability model for each basic event. See Note DA-N-2	
DA-A4	IDENTIFY the parameter to be estimated and the data required for estimation. See Note DA-N-3	
DA-A5	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives for Data Analysis in a manner that supports the applicable requirements of HLR-ESQ-E.	
DA-A6	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Data Analysis. See Note DA-N-4, DA-N-5	

Table 4.3.7.1-3 Supporting Requirements for HLR-DA-B

Grouping components into a homogeneous population for parameter estimation shall consider the design, environmental, and service conditions of the components in the as-built and as-operated plant. (HLR-DA-B)

Index No. DA-B	Capability Category I	Capability Category II
DA-B1	For parameter estimation, GROUP components according to type (e.g., motor-operated pump, air-operated valve).	For parameter estimation, GROUP components according to type (e.g., motor-operated pump, air-operated valve) and according to the characteristics of their usage to the extent supported by data. *(a)* mission type (e.g., standby, operating); *(b)* service condition (e.g., clean versus untreated water, air, harsh environments). Additional grouping characteristics may also be considered.
DA-B2	DO NOT INCLUDE outliers in the definition of a group (e.g., do not group valves that are never tested or intended to be tested and unlikely to be operated with those that are tested, intended to be tested, or otherwise manipulated frequently).	

Table 4.3.7.1-4 Supporting Requirements for HLR-DA-C

Generic parameter estimates shall be chosen, and plant-specific data shall be collected consistent with the parameter definitions of the Supporting Requirements (SRs) of HLR-DA-A and the grouping rationale of the SRs of HLR-DA-B. (HLR-DA-C)

Index No. DA-C	Capability Category I	Capability Category II
DA-C1	COLLECT generic parameter estimates as applicable to each modeled plant operating state from recognized sources. ENSURE that the parameter definitions and boundary conditions are consistent with those established in response to Requirements DA-A1, DA-A2, and DA-A3. (Example: Some sources include the breaker within the pump boundary, whereas others do not.) DO NOT INCLUDE generic data for unavailability due to test, maintenance, and repair unless it can be established that the data are consistent with the test and maintenance philosophies for the subject plant. If the generic data were developed for other types of reactors, JUSTIFY the applicability of the data to the specific reactor type and plant design. JUSTIFY use of the same parameter estimates in different plant operating states. See Note DA-N-6, DA-N-7, DA-N-8	
DA-C2	For PRAs performed during the pre-operational stage, INCLUDE relevant, available, and applicable experience from other facilities (potentially including nonnuclear facilities, fuel cycle facilities, and nonpower reactors) in the quantification of data parameters. See Note DA-N-5	
DA-C3	For operating plants, COLLECT plant-specific data for the basic event/parameter grouping corresponding to that defined by Requirements DA-A1, DA-A2, DA-A3, DA-B1, and DA-B2. See Note DA-N-9, DA-N-10, DA-N-11, DA-N-12, DA-N-13	
DA-C4	For operating plants, COLLECT plant-specific data, consistent with uniformity in design, operational practices, and experience. JUSTIFY the rationale for excluding plant-specific data (e.g., plant design modifications, changes in operating practices). See Note DA-N-10, DA-N-12, DA-N-13	

Table 4.3.7.1-4 Supporting Requirements for HLR-DA-C (Cont'd)

Generic parameter estimates shall be chosen, and plant-specific data shall be collected consistent with the parameter definitions of the Supporting Requirements (SRs) of HLR-DA-A and the grouping rationale of the SRs of HLR-DA-B. (HLR-DA-C)

Index No. DA-C	Capability Category I	Capability Category II
DA-C5	For operating plants, when evaluating maintenance or other relevant records to extract plant-specific component failure event data, SPECIFY a basis for the identification of events as failures. DELINEATE between those degraded states for which a failure, as modeled in the PRA, would have occurred during the mission and those for which a failure would not have occurred (e.g., slow pickup to rated speed). INCLUDE all failures that would have resulted in failure to perform the mission as defined in the PRA. See Note DA-N-10, DA-N-13	
DA-C6	For operating plants, COUNT repeated plant-specific component failures occurring within a short time interval as a single failure if there is a single, repetitive problem that causes the failures. In addition, COUNT only one demand. See Note DA-N-13	
DA-C7	For operating plants, ESTIMATE the number of plant-specific demands on standby components on the basis of the number of *(a)* surveillance tests; *(b)* maintenance acts; *(c)* surveillance tests or maintenance on other components; *(d)* operational demands. DO NOT COUNT additional demands from post-maintenance testing; that is part of the successful renewal. See Note DA-N-10, DA-N-13, DA-N-14	
DA-C8	For operating plants, EVALUATE the number of demands based upon the annualized number of surveillance tests and planned maintenance activities per plant procedures. See Note DA-N-13, DA-N-15	For operating plants, EVALUATE the number of demands based upon actual practice, including plant surveillance and maintenance tests, surveillances required by Technical Specification action statements, plant logs, etc. BASE the number of planned maintenance activities on plant maintenance plans and actual practice. BASE the number of unplanned maintenance acts on actual plant experience. See Note DA-N-13, DA-N-15
DA-C9	For PRAs performed during the pre-operational stage, ESTIMATE the number of surveillance tests and planned maintenance activities on plant requirements. See Note DA-N-5, DA-N-15	For PRAs performed during the pre-operational stage, EVALUATE the number of demands based upon expected performance, including planned plant surveillance and maintenance intervals, surveillances required by Technical Specification action statements, etc. BASE the number of planned maintenance activities on projected plant maintenance plans. BASE the number of unplanned maintenance acts on records from comparable systems in either nuclear or non-nuclear facilities. See Note DA-N-5, DA-N-15

Table 4.3.7.1-4 Supporting Requirements for HLR-DA-C (Cont'd)

Generic parameter estimates shall be chosen, and plant-specific data shall be collected consistent with the parameter definitions of the Supporting Requirements (SRs) of HLR-DA-A and the grouping rationale of the SRs of HLR-DA-B. (HLR-DA-C)

Index No. DA-C	Capability Category I	Capability Category II
DA-C10	For operating plants, when required, EVALUATE the time that components were configured in their standby status. See Note DA-N-13, DA-N-15	For operating plants, when required, EVALUATE the time that components were configured in their standby status using plant-specific operational records. See Note DA-N-10, DA-N-13, DA-N-15
DA-C11	For operating plants, ESTIMATE operational time from surveillance test practices for standby components and from actual operational data. See Note DA-N-13, DA-N-15	
DA-C12	For operating plants, when using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. Include only completed tests or unplanned operational demands as success for component operations. See Note DA-N-10, DA-N-13	For operating plants, when using surveillance test data, REVIEW the test procedure to determine whether a test should be credited for each possible failure mode. Include only completed tests or unplanned operational demands as success for component operation. If the component failure mode is decomposed into subelements (or causes) that are fully tested, then USE tests that exercise specific subelements in their evaluation. Thus, one subelement sometimes has many more successes than another. See Note DA-N-10, DA-N-13, DA-N-16
DA-C13	For operating plants, when using data on maintenance and testing durations to estimate unavailabilities at the component, train, or system-level, as required by the system model, only INCLUDE those maintenance or test activities that could leave the component, train, or system unable to perform its function when demanded. See Note DA-N-13	
DA-C14	For PRAs performed during the pre-operational stage, JUSTIFY the use of generic data for test and maintenance unavailability parameters. See Note DA-N-5	
DA-C15	When an unavailability of a front-line system component is caused by an unavailability of a support system, ASSIGN the unavailability toward that of the support system and not the front-line system unavailability, to avoid double counting and to capture support system dependency properly.	
DA-C16	For operating plants, EVALUATE the duration of the actual time that the equipment was unavailable for each contributing activity. INCLUDE low power and shutdown (LPSD) plant operating state and special maintenance activities. If reliable estimates of the start and finish times of periods of unavailability are not available, provide conservative estimates. See Note DA-N-13, DA-N-15, DA-N-17, DA-N-18, DA-N-19	For operating plants, EVALUATE the duration of the actual time that the equipment was unavailable for each contributing activity. INCLUDE LPSD plant operating state and special maintenance activities. If reliable estimates of the start and finish times are not available, INTERVIEW the knowledgeable plant personnel (e.g., engineering, plant operations) to generate estimates for ranges in the unavailable time per maintenance act for components, trains, or systems for which the unavailabilities are risk-significant basic events. See Note DA-N-13, DA-N-15, DA-N-17, DA-N-18, DA-N-19

Table 4.3.7.1-4 Supporting Requirements for HLR-DA-C (Cont'd)

Generic parameter estimates shall be chosen, and plant-specific data shall be collected consistent with the parameter definitions of the Supporting Requirements (SRs) of HLR-DA-A and the grouping rationale of the SRs of HLR-DA-B. (HLR-DA-C)

Index No. DA-C	Capability Category I	Capability Category II
DA-C17	For PRAs performed during the pre-operational stage, as plant-specific operating experience will not be available, IDENTIFY assumptions and bases for the unavailability of equipment for maintenance so as to meet the intent of Capability Category I (CC-I) of DA-C16. See Note DA-N-5, DA-N-17	For PRAs performed during the pre-operational stage, as plant-specific operating experience will not be available, IDENTIFY assumptions and bases for the unavailability of equipment for maintenance so as to meet the intent of Capability Category II (CC-II) of DA-C16. See Note DA-N-5, DA-N-17
DA-C18	For operating plants, EVALUATE coincident unavailability due to maintenance for redundant equipment (both intrasystem and intersystem) that is a result of a planned, repetitive activity based on actual plant experience. CALCULATE coincident maintenance unavailabilities that are a result of a planned, repetitive activity that represents actual plant experience. See Note DA-N-15, DA-N-18, DA-N-20	
DA-C19	For PRAs performed during the pre-operational stage, plant-specific operating experience will not be available, IDENTIFY assumptions and bases for the unavailability of equipment for maintenance so as to meet the intent of DA-C18. See Note DA-N-5	
DA-C20	For each structure, system, and component for which repair is to be modeled (see Requirement SY-A31), IDENTIFY instances of plant-specific or applicable industry experience, and for each repair, COLLECT the associated repair time with the repair time being the period from identification of the component failure until the component is returned to service.	
DA-C21	For PRAs performed during the pre-operational stage, IDENTIFY assumptions and bases for applicable industry experience used to satisfy DA-C20. See Note DA-N-4, DA-N-5	
DA-C22	For operating plants, data on recovery from loss of off-site power (LOOP), loss of service water, etc., are rare on a plant-specific basis. If available, for each recovery, COLLECT the associated recovery time with the recovery time being the period from identification of the system or function failure until the system or function is returned to service. See Note DA-N-13, DA-N-21	
DA-C23	For PRAs performed during the pre-operational stage, USE generic data for off-site power recovery, as applicable. See Note DA-N-5	
DA-C24	For operating plants, COLLECT plant-specific outage timeline data, accounting for plant operating state start time and duration and special maintenance configurations for each LPSD evolution (see also Requirement POS-C1). See Note DA-N-13, DA-N-22	
DA-C25	EVALUATE the use of the same generic parameter estimates for multiple plant operating states. DO NOT USE the same generic estimates for multiple plant operating states unless it can be established that the generic estimates are applicable for all such plant operating states.	
DA-C26	For operating plants, COLLECT the number of outages per calendar-year, accounting for outage types during each LPSD evolution (see Requirement POS-C1). See Note DA-N-13	

Table 4.3.7.1-5 Supporting Requirements for HLR-DA-D

The parameter estimates shall be based on relevant generic industry and technology- and plant-specific evidence. Where feasible, generic and technology- and plant-specific evidence shall be integrated using acceptable methods to obtain technology- and plant-specific parameter estimates. Each parameter estimate shall be accompanied by a characterization of the uncertainty. (HLR-DA-D)

Index No. DA-D	Capability Category I	Capability Category II
DA-D1	USE plant- or technology-specific parameter estimates for basic events and plant operating states modeling the unique design or operational features if available, or use generic information modified as discussed in Requirement DA-D2; USE generic information for the remaining events. See Note DA-N-23, DA-N-24, DA-N-25	CALCULATE realistic parameter estimates for risk-significant basic events and risk-significant plant operating states based on relevant generic and available plant- and/or technology-specific evidence unless it is justified that there are adequate plant-specific data to characterize the parameter value and its uncertainty. When it is necessary to integrate evidence from generic and plant- and/or technology-specific data, USE a statistical process that assigns appropriate weight to the statistical significance of the generic and plant- and/or technology-specific evidence and provides a characterization of uncertainty. Use of either a non-informative prior or one that represents variability in industry data is acceptable. SELECT prior distributions as either noninformative or representative of variability in industry data. CALCULATE parameter estimates for the remaining events and plant operating states by using generic industry data. See Note DA-N-23, DA-N-24, DA-N-25
DA-D2	If neither plant-specific data, technology-specific data, nor generic parameter estimates are available for the parameter associated with a specific basic event, USE data or estimates for the most similar equipment available, adjusting if necessary, to account for differences. Alternatively, USE expert judgment, and document the rationale behind the choice of parameter values. If using expert judgment, SATISFY the requirements of Section 4.2, Use of Expert Judgment.	
DA-D3	CALCULATE a point estimate, and CHARACTERIZE the uncertainty for the basic event probabilities. See Note DA-N-23, DA-N-26	CALCULATE a mean value for the parameters used to calculate the probabilities of the risk-significant basic events. PROVIDE the probabilistic representation of the uncertainty of the parameter estimates of the risk-significant basic events. Acceptable methods include Bayesian updating or expert judgment. If using expert judgment, SATISFY the requirements of Section 4.2, Use of Expert Judgment. For the basic events that are not risk-significant, ENSURE the requirement for CC-I is met. See Note DA-N-23, DA-N-26
DA-D4	For PRAs on plants with operating experience, when the Bayesian approach is used to derive a distribution and mean value of a parameter, ENSURE that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the plant-specific data. See Note DA-N-13, DA-N-23, DA-N-27	

Table 4.3.7.1-5 Supporting Requirements for HLR-DA-D (Cont'd)

The parameter estimates shall be based on relevant generic industry and technology- and plant-specific evidence. Where feasible, generic and technology- and plant-specific evidence shall be integrated using acceptable methods to obtain technology- and plant-specific parameter estimates. Each parameter estimate shall be accompanied by a characterization of the uncertainty. (HLR-DA-D)

Index No. DA-D	Capability Category I	Capability Category II
DA-D5	For PRAs performed during the pre-operational stage, when the Bayesian approach is used to derive a distribution and mean value of a parameter, ENSURE that the posterior distribution is reasonable given the relative weight of evidence provided by the prior and the technology-specific data. See Note DA-N-5, DA-N-23, DA-N-24, DA-N-27	
DA-D6	ENSURE that equipment repair and recovery probabilities account for plant operating state and event sequence specific influences.	
DA-D7	USE the beta-factor approach or an equivalent for estimating common cause failure (CCF) parameters. See Note DA-N-28	USE one of the following models for estimating CCF parameters for risk-significant CCF basic events: *(a)* Alpha Factor Model; *(b)* Basic Parameter Model; *(c)* Multiple Greek Letter (MGL) Model; *(d)* Binomial Failure Rate Model. JUSTIFY the use of alternative methods (i.e., provide evidence of peer review or verification of the method that demonstrates its acceptability). For estimating CCF parameters for non-risk-significant basic events for CCF, ENSURE the requirement for CC-I is met.
DA-D8	USE generic CCF parameters. ENSURE the CCF parameters are evaluated in a manner consistent with the component boundaries. See Note DA-N-29	USE CCF parameters consistent with plant experience when available. ENSURE the CCF parameters are evaluated in a manner consistent with the component boundaries. See Note DA-N-29
DA-D9	If generic event data is excluded for plant-specific estimation, ENSURE that the generic event data is excluded on both the CCF events and the independent failure events in the database used to generate the CCF parameters.	
DA-D10	For operating plants, if modifications to plant design or operating practice lead to a condition where past data are no longer representative of current performance, LIMIT the use of old data *(a)* if the modification involves new equipment or a practice where generic parameter estimates are available, USE the generic parameter estimates updated with plant-specific data as they become available for unique design or operational features; or *(b)* if the modification is unique to the extent that generic parameter estimates are not available and only limited experience is available following the change, then ANALYZE the impact of the change, and ASSESS the hypothetical effect on the historical data to determine to what extent the data can be used. See Note DA-N-13, DA-N-30	For operating plants, if modifications to plant design or operating practice lead to a condition where past data are no longer representative of current performance, LIMIT the use of old data *(a)* if the modification involves new equipment or a practice where generic parameter estimates are available, USE the generic parameter estimates updated with plant-specific data as they become available for risk-significant basic events and unique design and operational features; or *(b)* if the modification is unique to the extent that generic parameter estimates are not available and only limited experience is available following the change, then ANALYZE the impact of the change, and ASSESS the hypothetical effect on the historical data to determine to what extent the data can be used. See Note DA-N-13, DA-N-30

Table 4.3.7.1-6 Supporting Requirements for HLR-DA-E

The documentation of the Data Analysis shall provide traceability of the work. (HLR-DA-E)

Index No. DA-E	Capability Category I / Capability Category II
DA-E1	DOCUMENT the process used in the Data Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* system and component boundaries used to establish component failure probabilities; *(b)* the model used to evaluate each basic event probability; *(c)* sources for generic parameter estimates; *(d)* the plant-specific and plant operating state-specific sources of data, including that used for plant operating state durations; *(e)* the time periods for which plant-specific data were gathered; justification of any censoring of the data for specific LPSD conditions; *(f)* justification for exclusion of any data; *(g)* the basis for the estimates of CCF probabilities, including justification for excluding or mapping of generic and plant-specific data; *(h)* the rationale for any distributions used as priors for Bayesian updates, where applicable; *(i)* parameter estimate including the characterization of uncertainty, as appropriate; *(j)* justification for use of full power or other plant operating state data; *(k)* rationale for using generic parameter estimates for one or more plant operating states and for using the same generic estimates for multiple plant operating states. See Note DA-N-31
DA-E2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement DA-A5) associated with the Data Analysis.
DA-E3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Data Analysis. See DA-A6 See Note DA-N-5

4.3.7.2 Peer Review Requirements for Data Analysis

4.3.7.2.1 Purpose

This Section provides requirements for peer review of the Data Analysis element of the PRA.

4.3.7.2.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the area of data analysis. The team members assigned to review the data analysis shall have experience specific to that area and the capability of recognizing plant-specific features of the analyses.

4.3.7.2.3 Review of Data Analysis to Confirm the Methodology

A review shall be performed on selected event sequences. The portion of the event sequences selected for review typically includes the following:

(a) data values and associated component boundary definitions for component failure modes (including those with high importance values) contributing to the risk calculated in the PRA;

(b) CCF values;

(c) the numerator and denominator for one data value for each major failure mode (e.g., failure to start, failure to run, and test and maintenance unavailabilities);

(d) equipment repair and recovery data.

4.3.7.3 References for Data Analysis

The following is a list of publications referenced in this Standard.

[DA-1] NUREG/CR-6823, "Handbook of Parameter Estimation for Probabilistic Risk Assessment," Sandia National Laboratories/U.S. Nuclear Regulatory Commission, September 2003

[DA-2] NUREG/CR-4639, "Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR)," Vols. 1–5, U.S. Nuclear Regulatory Commission, 1994

[DA-3] NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants," Idaho National Laboratory/U.S. Nuclear Regulatory Commission, February 2007

[DA-4] NUREG/CR-4550, Vol. 1, "Analysis of Core Damage Frequency from Internal Events: Methodology Guidelines," U.S. Nuclear Regulatory Commission, January 1990

[DA-5] NUREG-1715, "Component Performance Study — Turbine-Driven Pumps, 1987–1998, Commercial Power Reactors," Vol. 1; "Component Performance Study—Motor-Driven Pumps, 1987–1998, Commercial Power Reactors," Vol. 2; "Component Performance Study—Air-Operated Valves, 1987–1998, Commercial Power Reactors," Vol. 3; "Component Performance Study—Motor-Operated Valves, 1987–1998, Commercial Power Reactors," Vol. 4, U.S. Nuclear Regulatory Commission, 2020

[DA-6] NUREG/CR-5497, F. M. MARSHALL, D. M. RASMUSON, and A. MOSLEH, "Common-Cause Failure Parameter Estimations," U.S. Nuclear Regulatory Commission, 1998

[DA-7] NUREG/CR-6268, "Common-Cause Failure Database and Analysis System: Overview," Vol. 1; "Common-Cause Failure Database and Analysis System: Event Definition and Classification," Vol. 2; "Common-Cause Failure Database and Analysis System: Data Collection and Event Coding," Vol. 3; "Common-Cause Failure Database and Analysis System: Software Reference Manual," Vol. 4, U.S. Nuclear Regulatory Commission, 1998

[DA-8] NUREG/CR-5496, "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980–1996," U.S. Nuclear Regulatory Commission, 1998

[DA-9] NUREG/CR-5032, "Modeling Time to Recovery and Initiating Event Frequency for Loss-of-Off-Site Power Incidents at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, March 1988

[DA-10] NUREG/CR-5485, A. MOSLEH, D. M. RASMUSON, and F. M. MARSHALL, "Guidelines on Modeling Common-Cause Failures in Probabilistic Risk Assessment," U.S. Nuclear Regulatory Commission, November 20, 1998

NONMANDATORY APPENDIX DA: NOTES AND EXPLANATORY MATERIAL FOR DATA ANALYSIS

DA.1 NOTES ASSOCIATED WITH DATA ANALYSIS

Table DA-1 Notes Supporting Data Analysis Requirements

Number	Notes
DA-N-1	Examples of basic events include the following: *(a)* independent or CCF of a component or system to start or change state on demand; *(b)* independent or CCF of a component or system to continue operating or provide a required function for a defined time period; *(c)* equipment unavailable to perform its required function due to being out of service for maintenance; *(d)* equipment unavailable to perform its required function due to being in test mode; *(e)* failure to recover a function or system (e.g., failure to recover off-site power); *(f)* failure to repair a component, system, or function in a defined time period. See DA-A1
DA-N-2	Examples include the following: *(a)* $1 - e^{-\lambda t} = \lambda t$, when $\lambda t < 0.1$, for failure to continue running during mission time t with constant failure rate λ; *(b)* $\frac{1}{2}\lambda T$, for a periodically tested standby component subject to standby failure rate of λ and a testing interval of T. See DA-A3
DA-N-3	Examples are as follows: *(a)* for failures on demand, the parameter is the probability of failure, and the data required are the number of failures given a number of demands; *(b)* for standby failures, operating failures, and initiating events, the parameter is the failure rate, and the data required are the number of failures in the total (standby or operating) time; *(c)* for unavailability due to test or maintenance, the parameter is the unavailability on demand, and the alternatives for the data required include the following: *(1)* the total time of unavailability OR a list of the maintenance events with their durations, together with the total time required to be available; OR *(2)* the number of maintenance or test acts, their average duration, and the total time required to be available; *(d)* for plant operating state durations, the parameter is the duration of each plant operating state, and the data required are the durations for past evolutions; *(e)* for plant operating state frequencies, the parameter is the frequency plant operating states per evolutions, and the data required are the number of evolutions during the plant-calendar-year. See DA-A4
DA-N-4	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See DA-A6, DA-C21
DA-N-5	This SR is not applicable to operating plants. See DA-A6, DA-C2, DA-C9, DA-C14, DA-C17, DA-C19, DA-C21, DA-C23, DA-D5, DA-E3

Table DA-1 Notes Supporting Data Analysis Requirements (Cont'd)

Number	Notes
DA-N-6	Examples of parameter estimates and associated sources include the following: *(a)* component failure rates and probabilities: NUREG/CR-4639 [DA-2], NUREG/CR-6928 [DA-3], NUREG/CR-4550 [DA-4], NUREG-1715 [DA-5]; *(b)* CCFs: NUREG/CR-5497 [DA-6], NUREG/CR-6268 [DA-7]; *(c)* alternating-current off-site power recovery: NUREG/CR-5496 [DA-8], NUREG/CR-5032 [DA-9]; *(d)* component recovery. See DA-C1
DA-N-7	See NUREG/CR-6823 [DA-1] for a listing of additional data sources. See DA-C1
DA-N-8	This needs to be done on a plant operating state-specific basis. Use of the same estimates in multiple plant operating states requires care and justification. See DA-C1
DA-N-9	Generally, equipment failure data are no different during shutdown than during operations. However, several factors are important when considering using normal failure data. The following factors can affect all parameter estimates, not just equipment failure rates: *(a)* long plant evolutions with equipment far outside normal operating conditions and test practice can affect successful performance; *(b)* Systems Analysis can account for different test and operating practice during the plant evolution. See DA-C3
DA-N-10	Other Data Analysis SRs, including Requirements DA-C1 and DA-D2, provide needed and available data. See DA-C3, DA-C4, DA-C5, DA-C7, DA-C10, DA-C12
DA-N-11	Parameter estimates are affected by special configurations (reactor coolant system and maintenance) that occur during LPSD. See DA-C3
DA-N-12	The data collected and used should apply to the plant operating state being evaluated. This may include data from the specific plant operating state and any other plant operating states in which the equipment performance would be expected to be similar. Use of the same data in multiple plant operating states should be done with care and justification. See DA-C3, DA-C4
DA-N-13	This SR is likely not applicable to PRAs performed during the pre-operational stage. See DA-C3, DA-C4, DA-C5, DA-C6, DA-C7, DA-C8, DA-C10, DA-C11, DA-C12, DA-C13, DA-C16, DA-C22, DA-C24, DA-C26, DA-D4, DA-D10
DA-N-14	For LPSD evolutions, the counts may need to be specialized to LPSD conditions and even to specific shutdown maintenance conditions. The number of demands should apply to the plant operating state being evaluated. This may include the number of demands from the specific plant operating state and any other plant operating states in which the equipment performance would be expected to be similar. Use of the same data in multiple plant operating states requires care and justification. See DA-C7
DA-N-15	The data may need to be specialized to LPSD conditions and even to specific shutdown maintenance and plant operating state conditions. See DA-C8, DA-C9, DA-C10, DA-C11, DA-C16, DA-C18
DA-N-16	Example: A diesel generator is tested more frequently than the load sequencer. If the sequencer were to be included in the diesel generator boundary, the number of valid tests would be decreased. See DA-C12
DA-N-17	Maintenance duration estimates may need to be modified to account for LPSD conditions. See DA-C16, DA-C17

Table DA-1 Notes Supporting Data Analysis Requirements (Cont'd)

Number	Notes
DA-N-18	Note that out-of-service unavailability data are very different for shutdown conditions, primarily because of the following: *(a)* Equipment unavailabilities are correlated by planned maintenance configurations. *(b)* Equipment repair is more a function of outage schedule and outage management than actual time required to complete repair. Outage times may be much longer than at-power [i.e., there may be no limiting condition of operation or limiting condition of operation (LCO), and outage management considerations may defer restoration to service; thus, data for outage time are often to be based on policy and outage practice, rather than on past experience]. *(c)* Note that repair data can be very different for shutdown conditions, primarily because equipment repair is more a function of outage schedule and outage management than actual time required to complete repair. *(d)* Outage times may be much longer than at nominal full power (i.e., there may be no LCO, and outage management considerations may defer restoration to service; thus, data for outage time can often be based on policy and outage practice, rather than on past experience where full-power data are irrelevant to such cases). Realistic assessment of repair/restoration depends on a realistic assessment of LPSD conditions on a plant operating state by plant operating basis. Cognizance of outage planning considerations is essential. See DA-C16, DA-C18
DA-N-19	Special attention should be paid to the case of a multi-reactor plant with shared systems, when the Technical Specifications requirements can be different depending on the status of individual reactors. Accurate modeling generally leads to a particular allocation of outage data among basic events to take this mode dependence into account. See DA-C16
DA-N-20	Such coincident maintenance unavailability can arise, for example, for plant systems that have "installed spares," i.e., plant systems that have more redundancy than is addressed by Technical Specifications. For example (intrasystem case), the charging system in some plants has a third train that may be out of service for extended periods of time coincident with one of the other trains and yet is in compliance with Technical Specifications. Examples of intersystem unavailability include plants that routinely take out multiple components on a "train schedule." See DA-C18
DA-N-21	Note that other planned maintenance activities can have a major impact on recovery of off-site power outage, and plant operating state-specific corrections may be required. See DA-C22
DA-N-22	This requirement provides data not needed for full-power conditions. It is a function of the outage plan and uncertainties in the plant staff's ability to meet that plan. Thus, data collection may include the use of expert elicitation. Uncertainty information can be developed from timelines of previous outages combined with expert elicitation. All indications are that such data are very plant-specific and vary with time, especially in recent years. See DA-C24
DA-N-23	NUREG/CR-6823, "Handbook of Parameter Estimation for Probabilistic Risk Assessment" [DA-1], provides guidance. See DA-D1, DA-D3, DA-D4, DA-D5
DA-N-24	Technology-specific testing evidence includes testing data collected from relevant nuclear and non-nuclear testing programs for the component or system of interest. An example includes evidence collected from a separate effects test loop that will employ the same valve under approximately the same temperatures and working fluid conditions as modeled in the PRA. See DA-D1, DA-D5

Table DA-1 Notes Supporting Data Analysis Requirements (Cont'd)

Number	Notes
DA-N-25	This SR allows for the integration of generic data with plant- or technology-specific evidence. PRAs performed during the pre-operational stage will not have access to plant-specific data and will rely solely on technology-specific data to calculate realistic parameter estimates. Operational plants may use a combination of generic, plant-specific, and technology-specific data if each category of data is appropriately weighted to their statistical significance, which will likely change as operational experience is accrued. See DA-D1
DA-N-26	This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding. See DA-D3
DA-N-27	Examples of tests to ensure that the updating is accomplished correctly and that the generic parameter estimates are consistent with the plant-specific application include the following: *(a)* confirmation that the Bayesian updating does not produce a posterior distribution with a single-bin histogram; *(b)* examination of the cause of any unusual (e.g., multimodal) posterior distribution shapes; *(c)* examination of inconsistencies between the prior distribution and the plant-specific evidence to confirm that they are appropriate; *(d)* confirmation that the Bayesian updating algorithm provides meaningful results over the range of values being analyzed; *(e)* confirmation of the reasonableness of the posterior distribution mean value. See DA-D4, DA-D5
DA-N-28	See the screening approach in NUREG/CR-5485 [DA-10]. See DA-D7
DA-N-29	Note that equipment CCF data are a difficult area for LPSD conditions. Many of the underlying causes of CCF can be affected by physical activities during outages, changes in plant conditions, and outside personnel having access to plant equipment. Full-power CCF data may apply to the plant operating state and maintenance activities during each phase of LPSD. However, adjustments are often necessary. Cognizance of the many controls the plant has in place to keep workers from interacting with the “protected train” helps ensure that CCF probabilities are realistic. See DA-D8
DA-N-30	Unique design and operational features include features unique to non-LWR being analyzed and thus do not have readily available operational data. See DA-D10
DA-N-31	The documentation requirements illustrate that there is a record of how the special conditions that exist during LPSD are accounted for in the analysis. They provide a picture of the differences between plant operating states in the data and parameter estimation. See DA-E1

4.3.8 Internal Flood PRA (FL)

This Section presents the technical requirements associated with Internal Flood PRA.

The requirements in this Section are divided into the following technical subelements:

(a) Internal Flood Plant Partitioning (FLPP);
(b) Internal Flood Sources Identification and Characterization (FLSO);
(c) Internal Flood Scenarios Development (FLSN);
(d) Internal Flood Initiating Events (FLEV);
(e) Internal Flood Plant Response (FLPR);
(f) Internal Flood Human Reliability Analysis (FLHR);
(g) Internal Flood Event Sequence Quantification (FLESQ).

4.3.8.1 Objectives and Technical Requirements for Internal Flood Plant Partitioning (FLPP)

The objectives of Internal Flood Plant Partitioning ensure that

(a) plant-specific physical layouts and separations are included;
(b) flood areas are defined to provide the basis for the identification of flood scenarios and flood-induced event sequences; and
(c) the Internal Flood Plant Partitioning is documented to provide traceability of the analysis.

Table 4.3.8.1-1 High Level Requirements for Internal Flood Plant Partitioning

Designator	Requirement
HLR-FLPP-A	The Internal Flood PRA shall define the physical boundaries of the analysis so as to include all plant locations relevant to the Internal Flood PRA.
HLR-FLPP-B	The Internal Flood PRA shall perform a plant partitioning analysis to identify and define the flood areas to be evaluated in the Internal Flood PRA.
HLR-FLPP-C	Documentation of the Internal Flood Plant Partitioning shall provide traceability of the analysis.

Table 4.3.8.1-2 Supporting Requirements for HLR-FLPP-A

The Internal Flood PRA shall define the physical boundaries of the analysis so as to include all plant locations relevant to the Internal Flood PRA. (HLR-FLPP-A)

Index No. FLPP-A	Capability Category I	Capability Category II
FLPP-A1	INCLUDE within the plant analysis boundary all areas or locations within the licensee-controlled area where an internal flood could adversely affect any equipment to be included in the Internal Flood PRA plant response model, including those locations where a flood could impact two or more reactors or sources of radioactive material in the scope of the PRA.	

Table 4.3.8.1-3 Supporting Requirements for HLR-FLPP-B

The Internal Flood PRA shall perform a plant partitioning analysis to identify and define the flood areas to be evaluated in the Internal Flood PRA. (HLR-FLPP-B)

Index No. FLPP-B	Capability Category I	Capability Category II
FLPP-B1	DEFINE flood areas by dividing the plant into physically separate areas where a flood area is viewed as a portion of a building or plant that is separated from other areas by barriers that delay, restrict, or prevent the propagation of floods to adjacent areas. IDENTIFY changes in flood barriers, flood sources, flood propagation paths, and equipment susceptibilities due to changes in plant operating states within the scope of the PRA. See Note FL-N-1	
FLPP-B2	DEFINE flood areas in a manner that resolves the flood impacts on single or multiple reactors and/or sources of radioactive material, as applicable and consistent with the scope of the PRA. See Note FL-N-2	

Table 4.3.8.1-3 Supporting Requirements for HLR-FLPP-B (Cont'd)

The Internal Flood PRA shall perform a plant partitioning analysis to identify and define the flood areas to be evaluated in the Internal Flood PRA. (HLR-FLPP-B)

Index No. FLPP-B	Capability Category I	Capability Category II
FLPP-B3	For operating plants, USE plant information sources that represent the as-built, as-operated plant, to support development of flood areas. See Note FL-N-3	
FLPP-B4	For PRAs performed during the pre-operational stage, USE pertinent design information sources that appropriately represent the as-designed, as-intended-to-operate plant, to support development of flood areas. See Note FL-N-4	
FLPP-B5	ENSURE that *(a)* collectively, the defined flood areas encompass all locations within the plant analysis boundary, including areas where a flood may impact two or more reactors or sources of radioactive material within the scope of the PRA (see Requirement FLPP-A1); and *(b)* defined flood areas do not overlap.	
FLPP-B6	EVALUATE the Internal Flood Plant Partitioning for the as-built, as-operated or as-designed, as-intended-to-operate plant conditions via investigation(s) depending on the plant design-life cycle stage of the PRA to determine *(a)* spatial information needed for the development of flood areas; *(b)* plant design features credited in defining flood areas; and *(c)* assumptions incorporated about the design. See Note FL-N-5, FL-N-6	
FLPP-B7	IDENTIFY the sources of model uncertainty and related assumptions associated with plant partitioning in a manner that supports Requirement FLESQ-E1.	
FLPP-B8	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence plant partitioning. See Note FL-N-4, FL-N-7	

Table 4.3.8.1-4 Supporting Requirements for HLR-FLPP-C

Documentation of the Internal Flood Plant Partitioning shall provide traceability of the analysis. (HLR-FLPP-C)

Index No. FLPP-C	Capability Category I	Capability Category II
FLPP-C1	DOCUMENT the process used in the Internal Flood Plant Partitioning analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of Supporting Requirements (SR) are satisfied: *(a)* flood areas defined in the analysis and the reasons for excluding any areas within the licensee-controlled area from further analysis; *(b)* the general nature and key or unique features of the partitioning elements that define each flood area; *(c)* any investigation(s) performed in support of the plant partitioning; *(d)* impact of plant operating state changes within the scope of the PRA on plant partitioning. See Note FL-N-8	
FLPP-C2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FLPP-B7) associated with the Internal Flood Plant Partitioning analysis.	
FLPP-C3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details. See FLPP-B8. See Note FL-N-4	

4.3.8.2 Objectives and Technical Requirements for Internal Flood Sources Identification and Characterization (FLSO)

The objectives of Internal Flood Sources Identification and Characterization development are to define, screen, and document the plant-specific internal flood scenarios in such a way that

(a) systematic development of flood scenarios is performed; and

(b) the Internal Flood Sources Identification and Characterization development is documented to provide traceability of the analysis.

Table 4.3.8.2-1 High Level Requirements for Internal Flood Sources Identification and Characterization

Designator	Requirement
HLR-FLSO-A	The potential flood sources in the flood areas, and their associated failure mechanisms, shall be identified and characterized in a manner sufficient to define flood scenarios.
HLR-FLSO-B	Documentation of the Internal Flood Sources Identification and Characterization shall provide traceability of the analysis.

Table 4.3.8.2-2 Supporting Requirements for HLR-FLSO-A

The potential flood sources in the flood areas and their associated failure mechanisms shall be identified and characterized in a manner sufficient to define flood scenarios. (HLR-FLSO-A)

Index No. FLSO-A	Capability Category I	Capability Category II
FLSO-A1	For each flood area, IDENTIFY the potential flood sources, including the following: *(a)* equipment (e.g., piping, valves, pumps) located in the area that is connected to fluid systems; *(b)* plant internal flood sources located in the flood area; *(c)* plant external flood sources that are connected through some system or structure within the plant boundary; *(d)* plant operating state-specific flood sources within the scope of the PRA. See Note FL-N-9, FL-N-10	
FLSO-A2	IDENTIFY the potential flood sources that include water and steam.	
FLSO-A3	For multi-reactor plants with shared systems or structures, INCLUDE any sources with potential impacts on multiple reactors or sources of radioactive material. See Note FL-N-2	
FLSO-A4	RETAIN flood areas for further consideration as flood initiating areas unless it can be concluded, using criterion SCR-3 in Table 1.10-1, that they do not contain any of the potential flood sources identified via Requirements FLSO-A1, FLSO-A2, and FLSO-A3.	
FLSO-A5	For each potential flood source, IDENTIFY the failure mechanisms that would result in a release of water, steam, or other liquids from the flood source, including the following: *(a)* failure modes of components such as pipes, tanks, gaskets, expansion joints, fittings, seals, etc.; *(b)* human-induced mechanisms that could lead to overfilling tanks, diversion of flow through openings created to perform maintenance; *(c)* inadvertent actuation of a fire suppression system; *(d)* other events resulting in a release into the flood area. See Note FL-N-11	
FLSO-A6	For each source and its identified failure mechanism, IDENTIFY the characteristic of release and the capacity of the source, including the following: *(a)* a characterization of the breach, including flood type; *(b)* applicable range of flow rates; *(c)* capacity of source (e.g., gallons of water); *(d)* the pressure and temperature of the source. See Note FL-N-12	

Table 4.3.8.2-2 Supporting Requirements for HLR-FLSO-A (Cont'd)

The potential flood sources in the flood areas and their associated failure mechanisms shall be identified and characterized in a manner sufficient to define flood scenarios. (HLR-FLSO-A)

Index No. FLSO-A	Capability Category I	Capability Category II
FLSO-A7	IDENTIFY the location of flood sources and the possibility of flooding of the area due to inleakage pathways from plant information sources or via plant walkdown(s). CONFIRM the accuracy of information collected from plant information sources for the as-designed, or as-built, or as-operated and as-intended-to-operate plant conditions via investigation(s) depending on the plant design-life cycle stage of the PRA to determine the location of flood sources and the possibility of flooding of the area due to inleakage pathways. See Note FL-N-5	
FLSO-A8	IDENTIFY the sources of model uncertainty and related assumptions associated with flood source identification and characterization in a manner that supports Requirement FLESQ-E1.	
FLSO-A9	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence flood source identification and characterization. See Note FL-N-4, FL-N-7	

Table 4.3.8.2-3 Supporting Requirements for HLR-FLSO-B

Documentation of the Internal Flood Sources Identification and Characterization shall provide traceability of the analysis. (HLR-FLSO-B)

Index No. FLSO-B	Capability Category I	Capability Category II
FLSO-B1	DOCUMENT the process used in the internal flood source identification and characterization specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of Supporting Requirements (SRs) are satisfied: *(a)* identified flood sources and the resulting list of sources to be further examined; *(b)* identified failure mechanisms and flood characteristics; *(c)* basis for any screening performed; *(d)* any calculations or other analyses used to support or refine the flooding evaluation; *(e)* any investigations performed; *(f)* impact of plant operating state changes within the scope of the PRA on flood source characterization.	
FLSO-B2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FLSO-A8) associated with the internal flood source identification and characterization.	
FLSO-B3	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details. See FLSO-A9 See Note FL-N-4	

4.3.8.3 Objectives and Technical Requirements for Internal Flood Scenarios Development (FLSN)

The objectives of Internal Flood Scenarios Development are to define, screen, and document the plant-specific internal flood scenarios in such a way that

(a) systematic identification of flood scenarios is performed; and
(b) the Internal Flood Scenarios Development is documented to provide traceability of the analysis.

Table 4.3.8.3-1 High Level Requirements for Internal Flood Scenarios Development

Designator	Requirement
HLR-FLSN-A	The flood scenarios shall be developed and characterized for each flood source in each retained flood area by identifying the propagation path(s) of the source and the affected systems, structures, and components (SSCs).
HLR-FLSN-B	Documentation of the Internal Flood Scenarios Development shall provide traceability of the analysis.

Table 4.3.8.3-2 Supporting Requirements for HLR-FLSN-A

The flood scenarios shall be developed and characterized for each flood source in each retained flood area by identifying the propagation path(s) of the source and the affected SSCs. (HLR-FLSN-A)

Index No. FLSN-A	Capability Category I	Capability Category II
FLSN-A1	IDENTIFY the flood propagation path from the flood source to its area(s) of accumulation for each flood source in each flood area retained in the internal flood PRA. See Note FL-N-1	
FLSN-A2	IDENTIFY plant design features that support the ability to terminate or contain the flood propagation for each flood source in each flood area retained in the internal flood PRA. INCLUDE the presence of the following, if applicable: *(a)* flood alarms; *(b)* flood dikes, curbs, sumps, water-tight doors, and all other flood barriers; *(c)* drains (i.e., physical structures that can function as drains); *(d)* sump pumps; *(e)* spray shields; and *(f)* blowout panels or dampers with automatic or manual operation capability. See Note FL-N-13	
FLSN-A3	IDENTIFY those automatic actuations or operator responses that have the ability to terminate or contain the flood propagation for each flood source in each flood area retained in the internal flood PRA. See Note FL-N-14	
FLSN-A4	ESTIMATE the capacity of the drains and the amount of water retained by sumps, berms, dikes, and curbs. INCLUDE these factors in estimating flood volumes and evaluating SSC impacts from flooding.	
FLSN-A5	IDENTIFY the following SSCs located in the flood area retained in the internal flood PRA, the spatial location of each identified SSC in the area, and any credited flooding mitigative features for each identified SSC that *(a)* are required to respond to an internal flood initiating event or whose failure would challenge normal plant operation, and are susceptible to flood; and *(b)* impact the ability to terminate, delay, or contain the flood propagation. See Note FL-N-15	

Table 4.3.8.3-2 Supporting Requirements for HLR-FLSN-A (Cont'd)

The flood scenarios shall be developed and characterized for each flood source in each retained flood area by identifying the propagation path(s) of the source and the affected SSCs. (HLR-FLSN-A)

Index No. FLSN-A	Capability Category I	Capability Category II
FLSN-A6	For the SSCs identified in Requirement FLSN-A5, IDENTIFY the susceptibility of each SSC in a flood area to submergence and spray failure mechanisms. Either ASSESS qualitatively the impact of the flood-induced mechanisms that are not explicitly addressed or SPECIFY that these mechanisms are not included in the scope of the evaluation. See Note FL-N-16	For the SSCs identified in Requirement FLSN-A5, IDENTIFY *(a)* the susceptibility of each SSC in a flood area to submergence, spray, humidity, and condensation failure mechanisms; *(b)* the susceptibility of each SSC to submergence, spray, jet impingement, pipe whip, temperature, pressure, humidity, and condensation failure mechanisms for flood scenarios involving a high energy line break. JUSTIFY any determination that SSCs as identified in Requirement FLSN-A5 within the flood area are not susceptible to flood-induced failure mechanisms.
FLSN-A7	When determining the susceptibility of SSCs to flood-induced failure mechanisms (see Requirement FLSN-A6), INCLUDE the operability of SSCs identified in Requirement FLSN-A5 unless the SSC functionality in the presence of internal flood effects can be supported by one or an appropriate combination of the following: *(a)* test or operational data; *(b)* engineering analysis; and *(c)* expert judgment (SATISFY the requirements of Section 4.2, Use of Expert Judgment).	
FLSN-A8	IDENTIFY inter-area propagation between areas connected via permanent opening(s); drain lines in the normal flow path; and open doors, stairwells, and hatchways. See Note FL-N-13	IDENTIFY inter-area propagation between areas connected via the following: *(a)* drain lines in the normal flow path; *(b)* backflow through drain lines involving failed check valves; *(c)* pipe and cable penetrations (including cable trays) without penetration seals; *(d)* doors and gaps below doors; *(e)* stairwells; *(f)* hatchways; *(g)* blow-out panels; *(h)* heating, ventilation, and air-conditioning (HVAC) ducts; *(i)* floor grates and plugs; and *(j)* penetration seals. INCLUDE potential for structural failure (e.g., doors, walls, penetration seals) due to flooding loads. See Note FL-N-13

Table 4.3.8.3-2 Supporting Requirements for HLR-FLSN-A (Cont'd)

The flood scenarios shall be developed and characterized for each flood source in each retained flood area by identifying the propagation path(s) of the source and the affected SSCs. (HLR-FLSN-A)

Index No. FLSN-A	Capability Category I	Capability Category II
FLSN-A9	For each flood scenario, using conservative plant- or design-specific values for flood area design features, ESTIMATE the following except where the requirements are not applicable: *(a)* conservative (i.e., bounding) flood source inventory, break size, and release rate; *(b)* conservative flood propagation and drainage rates; *(c)* conservative volume fractions occupied by SSCs for the affected flood areas (for flood-submergence scenarios only); *(d)* conservative potential of flood barrier failures; *(e)* conservative humidity and temperature conditions for the affected flood areas (for steam release scenario only). See Note FL-N-17	For each flood scenario that is risk-significant, using plant- or design-specific values for flood area design features, CALCULATE the following except where the requirements are not applicable: *(a)* flood source inventory, break size, and release rate; *(b)* flood propagation and drainage rates by including flow pathways through floor drains, floor grates, floor hatches, gaps below doorways, wall openings, and HVAC ducts; *(c)* SSC occupancy fractions for the affected flood areas (for flood-submergence scenarios only); *(d)* potential of flood barrier failures; *(e)* humidity and temperature conditions for the affected flood areas (for steam release scenarios only). See Note FL-N-17
FLSN-A10	For each flood scenario that causes submergence, ESTIMATE the maximum flood heights and the associated times to damage SSCs that are included in the internal flood PRA) model and are located in the flood initiating area and areas in potential propagation paths. ASSESS the impact on SSCs included in the Internal Flood PRA model caused by submergence, spray, harsh environment, or hydraulic loading in the flood initiating area and areas in potential propagation paths. See Note FL-N-1	For each flood scenario that causes submergence and is risk-significant, CALCULATE the flood heights and the associated times to damage SSCs that are included in the internal flood PRA) model and are located in the flood initiating area and areas in potential propagation paths. ASSESS the impact on SSCs included in the Internal Flood PRA model caused by submergence, spray, harsh environment, or hydraulic loading in the flood initiating area and areas in potential propagation paths. See Note FL-N-1
FLSN-A11	In the calculation of flood height in each flood area for each flood scenario that causes submergence, ENSURE that the propagation flow rates used do not result in non-conservative flood height for either the originating flood area (outleakage flow rate) or the receiving flood area (inleakage flow rate) along the propagation path.	
FLSN-A12	ENSURE that a realistic estimate of duration is used in the flood height analysis for each flood scenario that causes submergence so that the maximum flood height or critical flood height for susceptible equipment in each flood area along the flood propagation path (including the flood initiating area) is reached.	
FLSN-A13	DEVELOP flood scenarios by examining the equipment and relevant plant features in the flood area and areas in potential propagation paths, giving credit for appropriate flood mitigation systems or operator actions identified in Requirement FLSN-A3 and identifying susceptible SSCs. INCLUDE in the development of scenarios, the flood area, flood source, flood rate, flood propagation path, possibility of flood barrier failure, flood impact on plant SSCs, and human actions considered in flood initiation, mitigation, and termination. See Note FL-N-1, FL-N-18, FL-N-19	

Table 4.3.8.3-2 Supporting Requirements for HLR-FLSN-A (Cont'd)

The flood scenarios shall be developed and characterized for each flood source in each retained flood area by identifying the propagation path(s) of the source and the affected SSCs. (HLR-FLSN-A)

Index No. FLSN-A	Capability Category I	Capability Category II
FLSN-A14	INCLUDE flood scenarios that impact multiple reactors or sources of radioactive material, as applicable. See Note FL-N-20	
FLSN-A15	IDENTIFY the specific combination of reactors or sources of radioactive material affected by the flood scenario. See Note FL-N-20	
FLSN-A16	ASSESS the impact of fluids that may be chemically or physically incompatible. See Note FL-N-21	
FLSN-A17	RETAIN flood areas unless it can be concluded, using criterion SCR-3 in Table 1.10-1, that flooding of the flood area does not cause a flood-induced initiating event or a need for immediate plant shutdown, and any of the following applies: *(a)* the flood area (including adjacent areas where flood sources can propagate) contains no equipment modeled in the PRA or contains no equipment that supports the function of the modeled equipment; *(b)* the flood area has no flood sources sufficient to cause failure (e.g., through spray, immersion, or other applicable cause) of the equipment identified in Requirement FLSN-A5 (including equipment in adjacent areas where floods may propagate); *(c)* Requirement FLSN-A18 is met for all flood sources within that flood area. ENSURE that the failure of a barrier resulting in inter-area propagation is not used to justify screening out of flood areas (i.e., do not credit such failures as a means of beneficially draining the area without justification). See Note FL-N-22	
FLSN-A18	For flood areas retained via Requirement FLSN-A17, RETAIN flood sources unless it is concluded, using criterion SCR-3 in Table 1.10-1, that flooding of the flood area, based on the limiting flood defined for that source, does not cause an initiating event nor a need for immediate plant shutdown due to loss of function of one or more SSCs caused by the flood, and each of the following applies: *(a)* the flood area contains flood mitigation systems capable of preventing unacceptable flood levels; *(b)* the nature of the limiting flood does not cause failure of the flood mitigation systems or SSCs that are needed to prevent or mitigate an event sequence due to a flood-induced failure mechanism; and *(c)* there is no propagation to another flood area. ENSURE that mitigation systems are not used for screening out flood sources unless there is a basis for crediting the capability and reliability of the flood mitigation system(s). See Note FL-N-22, FL-N-23	
FLSN-A19	EVALUATE the accuracy of information collected from plant information sources for the as-designed, or as-built, and as-operated or as-intended-to-operate plant conditions via investigation(s) depending on the plant design-life cycle stage of the PRA to CONFIRM the following information: *(a)* SSCs located within each defined flood area; *(b)* flood/spray/other applicable mitigative features of the SSCs located within each defined flood area; and *(c)* flood propagation paths. See Note FL-N-5	
FLSN-A20	IDENTIFY the sources of model uncertainty associated with flood scenario development in a manner that supports Requirement FLESQ-E1.	
FLSN-A21	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence flood scenario development. See Note FL-N-4, FL-N-7	

Table 4.3.8.3-3 Supporting Requirements for HLR-FLSN-B

Documentation of the Internal Flood Scenarios Development shall provide traceability of the analysis. (HLR-FLSN-B)

Index No. FLSN-B	Capability Category I	Capability Category II
FLSN-B1	DOCUMENT the process used in the Internal Flood Scenarios Development specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* flood propagation paths and assumptions, calculations, or other bases for identifying and eliminating them; *(b)* event sequence-mitigating features and barriers credited in the analysis and associated justification; *(c)* flood scenarios considered, screened out, and retained; *(d)* screening criteria used in the analysis; *(e)* assumptions, justifications, and calculations used in the determination of flood-induced failure mechanisms (e.g., justification for the nonsusceptibility of SSCs to flood-induced failure mechanisms for modeled flood scenarios); *(f)* calculations or other analyses used to support or refine the flooding evaluation; *(g)* investigation(s) performed; *(h)* impact of plant operating state changes within the scope of the PRA on flood scenario characterization.	
FLSN-B2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FLSN-A20) associated with the Internal Flood Scenarios Development.	
FLSN-B3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details. See FLSN-A21 See Note FL-N-4	

4.3.8.4 Objectives and Technical Requirements for Internal Flood Initiating Events (FLEV)

The objectives of Internal Flood Initiating Events are to identify, quantify, and document the applicable flood-induced initiating event for each flood scenario that could lead to radionuclide release in such a way that

(a) internal flood-induced initiating events that challenge normal plant operation and that require successful mitigation to prevent radionuclide release are included;

(b) internal flood-induced initiating events and frequencies are grouped according to mitigation requirements to facilitate the efficient modeling of plant response;

(c) the internal flood-induced initiating events are documented to provide traceability of the analysis.

Table 4.3.8.4-1 High Level Requirements for Internal Flood Initiating Events

Designator	Requirement
HLR-FLEV-A	The Internal Flood Initiating Events shall identify flood-induced initiating events to be evaluated in the Internal Flood PRA plant response model.
HLR-FLEV-B	The Internal Flood Initiating Events shall quantify the annual frequencies of scenarios resulting in flood-induced initiating events.
HLR-FLEV-C	Documentation of the Internal Flood Initiating Events shall provide traceability of the analysis.

Table 4.3.8.4-2 Supporting Requirements for HLR-FLEV-A

The Internal Flood Initiating Events shall identify flood-induced initiating events to be evaluated in the Internal Flood PRA plant response model. (HLR-FLEV-A)

Index No. FLEV-A	Capability Category I	Capability Category II
FLEV-A1	GROUP flood scenarios identified in Requirement FLSN-A13 only when *(a)* flood scenarios can be considered similar in terms of plant operating state, plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or *(b)* flood scenarios are bounded by the worst-case impacts within the group.	GROUP flood scenarios identified in Requirement FLSN-A13 only when *(a)* flood scenarios can be considered similar in terms of plant operating state, plant response, success criteria, timing, and the effect on the operability and performance of operators and relevant mitigating systems; or *(b)* flood scenarios can be bounded by the worst-case impacts, including radionuclide release potential, within the group and the grouping does not impact identification of risk-significant event sequences.
FLEV-A2	For each flood scenario or flood-scenario group defined according to Requirement FLEV-A1, IDENTIFY the corresponding initiating event group from the internal events PRA. If an appropriate initiating event or initiating event group does not exist, CREATE a new initiating event group and SATISFY the Capability Category I (CC-I) requirements IE-A7, IE-A11, IE-A12, and IE-A13 for Initiating Events Analysis. See Note FL-N-24	For each flood scenario or flood-scenario group defined according to Requirement FLEV-A1, IDENTIFY the corresponding initiating event group from the internal events PRA. If an appropriate initiating event or initiating event group does not exist, CREATE a new initiating event group and SATISFY the Capability Category II (CC-II) requirements IE-A7, IE-A11, IE-A12, and IE-A13 for Initiating Event Analysis. See Note FL-N-24
FLEV-A3	For multi-reactor plants with shared systems or structures, INCLUDE multi-reactor impacts on SSCs in the definition and grouping of flood-induced initiating events. See Note FL-N-2	
FLEV-A4	IDENTIFY the specific combination of reactors or sources of radioactive material in shared facilities associated with the flood-induced initiating events. See Note FL-N-20	

Table 4.3.8.4-3 Supporting Requirements for HLR-FLEV-B

The Internal Flood Initiating Events shall quantify the annual frequencies of scenarios resulting in flood-induced initiating events. (HLR-FLEV-B)

Index No. FLEV-B	Capability Category I	Capability Category II
FLEV-B1	If choosing to include in the flood scenario definition mitigating features that have the ability to terminate or contain the flood propagation, QUANTIFY their probabilities of failure and SATISFY the CC-I SRs of HLR-SY-A and HLR-SY-B for Systems Analysis, as well as SRs of HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D for Data Analysis except where the requirements are not applicable.	If choosing to include in the flood scenario definition mitigating features that have the ability to terminate or contain the flood propagation, QUANTIFY their probabilities of failure and SATISFY the CC-II SRs of HLR-SY-A and HLR-SY-B for Systems Analysis, as well as SRs of HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D for Data Analysis except where the requirements are not applicable.
FLEV-B2	QUANTIFY the frequency for each flood-induced initiating event or initiating event group on a plant-year basis and INCLUDE the probability of failure of any mitigating features (e.g., Requirement FLEV-B1) and/or human error probabilities (HEPs) (e.g., Requirement FLHR-C1) that have been used to define the flood scenario and the associated flood-induced initiating event. See Note FL-N-11	
FLEV-B3	In estimating the flood-induced initiating event frequencies, USE one or a combination of the following: *(a)* generic operating experience; *(b)* pipe, component, and tank rupture failure rates from generic data sources; *(c)* expert judgment (SATISFY the requirements of Section 4.2, Use of Expert Judgment). See Note FL-N-24, FL-N-25	In estimating the flood-induced initiating event frequencies, USE plant- or design-specific information on operating practices and conditions that may impact flood-induced initiating event frequency (e.g., material condition of fluid systems, experience with water hammer, and maintenance-induced floods). For risk-significant pipe failure modes, INCLUDE pipe age-dependent failure rates where appropriate and when supported by applicable generic or plant- or design-specific data. In estimating the flood-induced initiating event frequencies, USE the above collected plant- or design-specific information and one or a combination of the following: *(a)* generic and plant-specific operating experience; *(b)* pipe, component, and tank rupture failure rates from generic data sources and plant-specific experience; *(c)* expert judgment for consideration of the plant-specific information collected (SATISFY the requirements of Section 4.2, Use of Expert Judgment). See Note FL-N-24, FL-N-25
FLEV-B4	ESTIMATE the frequency of human-induced floods during maintenance in each plant operating state through application of one of the following options: *(a)* available generic data; *(b)* available plant-specific data; *(c)* use of expert judgment (SATISFY the requirements of Section 4.2, Use of Expert Judgment); or *(d)* evaluation of HFEs during maintenance activities that can lead to human-induced floods and meet the CC-I SRs of HLR-HR-A, HLR-HR-B, HLR-HR-C, and HLR-HR-D except where the requirements are not applicable.	ESTIMATE the frequency of human-induced floods during maintenance in each plant operating state through the application of available generic and/or plant-specific data, or by using human reliability techniques in evaluating plant-specific maintenance activities. EVALUATE the HFEs during maintenance activities that can lead to human-induced floods and SATISFY the CC-II SRs of HLR-HR-A, HLR-HR-B, HLR-HR-C, and HLR-HR-D except where the requirements are not applicable.

Table 4.3.8.4-3 Supporting Requirements for HLR-FLEV-B (Cont'd)

The Internal Flood Initiating Events shall quantify the annual frequencies of scenarios resulting in flood-induced initiating events. (HLR-FLEV-B)

Index No. FLEV-B	Capability Category I / Capability Category II
FLEV-B5	RETAIN flood-induced initiating events or initiating event groups unless it can be concluded that Requirements IE-C9 can be satisfied, or any of the following items are satisfied: *(a)* SCR-1 or SCR-2 in Table 1.10-1, as applied to the flood-induced initiating events or initiating event groups, is directly met for affected event sequence families; or *(b)* the flood-induced initiating event affects only components in a single system, and it can be shown that the product of the frequency of the flood-induced initiating event and the probability of SSC failure given the flood is two orders of magnitude lower than the product of the non-flooding frequency for the corresponding initiating event in the PRA and the random (non-flood-induced) failure probability of the same SSCs that are assumed failed by the flood.
FLEV-B6	IDENTIFY the sources of model uncertainty and related assumptions associated with Internal Flood Initiating Events in a manner that supports Requirement FLESQ-E1.
FLEV-B7	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Internal Flood Initiating Events. See Note FL-N-4, FL-N-7

Table 4.3.8.4-4 Supporting Requirements for HLR-FLEV-C

Documentation of the Internal Flood Initiating Events shall provide traceability of the analysis. (HLR-FLEV-C)

Index No. FLEV-C	Capability Category I / Capability Category II
FLEV-C1	DOCUMENT the process used in the Internal Flood Initiating Events specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* basis for grouping and subsuming flood initiating events; *(b)* derivation of flood initiating event frequencies; *(c)* component unreliabilities/unavailabilities and HEPs used in the analysis (i.e., the data values unique to the internal flood analysis); *(d)* any calculations or other analyses used to support or refine the flooding evaluation; *(e)* basis for screened out flood-induced initiating events not impacting risk insights consistent with defined risk significance criteria; *(f)* impact of plant operating state changes within the scope of the PRA on flood-induced initiating events. Meet Requirement DA-E1 for Data Analysis, Requirement SY-C1 for System Analysis, and Requirement IE-D1 for Initiating Event Analysis except where the requirements are not applicable.
FLEV-C2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FLEV-B6) associated with the Internal Flood Initiating Events.
FLEV-C3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details. See FLEV-B7 See Note FL-N-4

4.3.8.5 Objectives and Technical Requirements for Internal Flood Plant Response (FLPR)

The objectives of the Internal Flood Plant Response model are to develop the internal flood-induced event sequences and the associated system, data, and human reliability analyses in a way such that

(a) the Internal Flood Plant Response model is capable of SRs in Internal Flood Event Sequence Quantification;

(b) all of the internal flood-induced initiating events identified are included; risk-significant event sequences for each internal flood-induced initiating event are included; risk-significant contributors including operator actions, mitigation systems, and phenomena that can alter internal flood event sequences are included in the event sequence model; plant-specific dependencies are represented in the event sequences; end states are clearly defined; the Internal Flood Plant Response model provides the basis for the quantification of the event sequences that may result from the internal flood scenarios and for the identification of the event sequence cutsets and risk-significant contributors;

(c) the Internal Flood Plant Response model development is documented to provide traceability of the analysis.

Table 4.3.8.5-1 High Level Requirements for Internal Flood Plant Response

Designator	Requirement
HLR-FLPR-A	The internal flood PRA shall include the Internal Flood Plant Response model capable of SRs in HLR-FLESQ-A, HLR-FLESQ-B, HLR-FLESQ-C, HLR-FLESQ-D, HLR-FLESQ-E, and HLR-FLESQ-F.
HLR-FLPR-B	The Internal Flood Plant Response model shall include flood-induced initiating events, both flood-induced and random failures of equipment, flood-specific as well as non-flood-related human failures associated with safe shutdown, radionuclide transport barrier failure modes, and the supporting probability data (including uncertainty) based on the SRs stated under this High Level Requirement (HLR) that parallel, as appropriate, the requirements of this Standard for internal events PRA.
HLR-FLPR-C	Documentation of the Internal Flood Plant Response model shall provide traceability of the analysis.

Table 4.3.8.5-2 Supporting Requirements for HLR-FLPR-A

The Internal Flood PRA shall include the Internal Flood Plant Response model capable of SRs in HLR-FLESQ-A, HLR-FLESQ-B, HLR-FLESQ-C, HLR-FLESQ-D, HLR-FLESQ-E, and HLR-FLESQ-F. (HLR-FLPR-A)

Index No. FLPR-A	Capability Category I	Capability Category II
FLPR-A1	CONSTRUCT the Internal Flood Plant Response model so that it is capable of determining the conditional probability of each flood-induced event sequence and event sequence family modeled in the PRA for the Internal Flood Scenarios Development, and their associated flood-induced impacts on mitigating equipment operator actions.	
FLPR-A2	CONSTRUCT the Internal Flood Plant Response model so that it is capable of determining the frequency of the flood-induced event sequences and event sequence families once the flood initiating event frequencies (see the SRs of HLR-FLEV-A and HLR-FLEV-B) are applied to the quantification.	
FLPR-A3	CONSTRUCT the Internal Flood Plant Response model so that it is capable of determining the risk-significant contributors consistent with Internal Flood Event Sequence Quantification.	

Table 4.3.8.5-3 Supporting Requirements for HLR-FLPR-B

The Internal Flood Plant Response model shall include flood-induced initiating events, both flood-induced and random failures of equipment, flood-specific as well as non-flood-related human failures associated with safe shutdown, radionuclide transport barrier failure modes, and the supporting probability data (including uncertainty) based on the SRs stated under this HLR that parallel, as appropriate, the requirements of this Standard for internal events PRA. (HLR-FLPR-B)

Index No. FLPR-B	Capability Category I	Capability Category II
FLPR-B1	USE the event sequences and the systems logic model from the internal event PRA models as the basis of the Internal Flood Plant Response model. INCLUDE additional event sequences, as applicable, associated with flood-induced initiating events that are not included in the internal events PRA model such as events that impact two or more reactors or sources of radioactive material. See IE-A16 See Note FL-N-26	
FLPR-B2	ENSURE that the peer review findings for the internal-events and other hazard PRAs that are relevant to the results of the internal flood PRA are resolved and incorporated into the development of the Internal Flood Plant Response model.	
FLPR-B3	For each flood scenario or flood-scenario group, REVIEW the event sequences for the associated initiating event group to confirm the applicability of the event sequence model. If appropriate event sequences do not exist, MODIFY existing event sequences or DEVELOP new sequences as necessary to include any unique event sequences that could result from the flood scenario and associated flood-induced failure mechanisms or phenomena. For the defined event sequences, meet the CC-I SRs of HLR-ES-A and HLR-ES-B for Event Sequence Analysis except where the requirements are not applicable.	For each flood scenario or flood-scenario group, REVIEW the event sequences for the associated initiating event group to confirm the applicability of the event sequence model. If appropriate event sequences do not exist, MODIFY existing event sequences or DEVELOP new sequences as necessary to include any unique event sequences that could result from the flood scenario and associated flood-induced failure mechanisms or phenomena. For the defined event sequences, meet the CC-II SRs of HLR-ES-A and HLR-ES-B for Event Sequence Analysis except where the requirements are not applicable.
FLPR-B4	MODEL event sequences for any new initiating events identified per Requirement FLEV-A2 that represent possible plant responses to the flood-induced initiating events and meet the SRs of HLR-ES-A and HLR-ES-B for the Event Sequence Analysis except where the requirements are not applicable with the following clarifications: *(a)* all the SRs under the CC-I of HLR-ES-A and HLR-ES-B are to be addressed in the context of internal flood scenarios; and *(b)* when applying Requirement ES-A5 to internal flood PRA, INCLUDE flood response procedures as well as emergency operating procedures and abnormal procedures.	MODEL event sequences for any new initiating events identified per Requirement FLEV-A2 that represent possible plant responses to the flood-induced initiating events and meet the SRs of HLR-ES-A and HLR-ES-B for the Event Sequence Analysis except where the requirements are not applicable with the following clarifications: *(a)* all the SRs under the CC-II of HLR-ES-A and HLR-ES-B are to be addressed in the context of internal flood scenarios; and *(b)* when applying Requirement ES-A5 to internal flood PRA, INCLUDE flood response procedures as well as emergency operating procedures and abnormal procedures.
FLPR-B5	IDENTIFY any cases where new or modified success criteria will be needed to support the internal flood PRA and SATISFY the CC-I SRs of HLR-SC-A for success criteria except where the requirements are not applicable.	IDENTIFY any cases where new or modified success criteria will be needed to support the internal flood PRA and SATISFY the CC-II SRs of HLR-SC-A for success criteria except where the requirements are not applicable.

Table 4.3.8.5-3 Supporting Requirements for HLR-FLPR-B (Cont'd)

The Internal Flood Plant Response model shall include flood-induced initiating events, both flood-induced and random failures of equipment, flood-specific as well as non-flood-related human failures associated with safe shutdown, radionuclide transport barrier failure modes, and the supporting probability data (including uncertainty) based on the SRs stated under this HLR that parallel, as appropriate, the requirements of this Standard for internal events PRA. (HLR-FLPR-B)

Index No. FLPR-B	Capability Category I	Capability Category II
FLPR-B6	DEFINE any new or modified success criteria identified per Requirement FLPR-B5 and MODEL the Internal Flood Plant Response by using these success criteria that meet the CC-I SRs of HLR-SC-B for success criteria except where the requirements are not applicable.	DEFINE any new or modified success criteria identified per Requirement FLPR-B5 and MODEL the Internal Flood Plant Response by using these success criteria that meet the CC-II SRs of HLR-SC-B for success criteria except where the requirements are not applicable.
FLPR-B7	MODIFY the existing systems models to include flood-induced failure mechanisms identified in accordance with Requirement FLSN-A6 or, if needed, PERFORM new Systems Analysis, and meet the CC-I SRs of HLR-SY-A and HLR-SY-B for Systems Analysis within the context of internal flood scenarios except where the requirements are not applicable. INCLUDE in the plant response model the effects of the following: *(a)* internal flood-induced equipment failures; and *(b)* internal flood-specific operator actions as identified per Internal Flood Human Reliability Analysis.	MODIFY the existing systems models to include flood-induced failure mechanisms identified in accordance with Requirement FLSN-A6 or, if needed, PERFORM new Systems Analysis, and meet the CC-II SRs of HLR-SY-A and HLR-SY-B for Systems Analysis within the context of internal flood scenarios except where the requirements are not applicable. INCLUDE in the plant response model the effects of the following: *(a)* internal flood-induced equipment failures; and *(b)* internal flood-specific operator actions as identified per Internal Flood Human Reliability Analysis.
FLPR-B8	REVIEW component mission times used in the plant response model for the flood scenarios retained to ensure that the impacts of the flood do not invalidate the assumed component mission time due to sustained impacts on the plant response. SATISFY the Requirement SC-A7 at CC-I for success criteria, except where the requirements are not applicable.	REVIEW component mission times used in the plant response model for the flood scenarios that are not qualitatively screened out to ensure that the impacts of the flood do not invalidate the assumed component mission time due to sustained impacts on the plant response. SATISFY the Requirement SC-A7 CC-II for success criteria, except where the requirements are not applicable.
FLPR-B9	IDENTIFY the sources of model uncertainty and related assumptions associated with the Internal Flood Plant Response model development in a manner that supports Requirement FLESQ-E1.	
FLPR-B10	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Internal Flood Plant Response model development. See Note FL-N-4, FL-N-7	

Table 4.3.8.5-4 Supporting Requirements for HLR-FLPR-C

Documentation of the Internal Flood Plant Response model shall provide traceability of the analysis. (HLR-FLPR-C)

Index No. FLPR-C	Capability Category I / Capability Category II
FLPR-C1	DOCUMENT the process used in the Internal Flood Plant Response model analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* description of the internal flood-induced initiating events and how the internal events PRA model were modified to model the internal flood-induced initiating events; *(b)* description of the success criteria established for each internal flood-induced initiating event including the bases for the criteria; *(c)* description of the internal flood-induced event sequence model developed for each internal flood-induced initiating event; *(d)* description of the revised and the new system analyses used to support the quantification of the internal flood-induced event sequence model; *(e)* description of the revised and new data analyses used to support the quantification of the internal flood-induced event sequence model; *(f)* impact of plant operating state changes within the scope of the PRA on plant response modeling. SATISFY SRs of HLR-IE-D for the Initiating Event Analysis, SRs of HLR-ES-D for the Event Sequence Analysis, SRs of HLR-SC-C for Success Criteria, SRs of HLR-SY-C for Systems Analysis, and SRs of HLR-DA-E for Data Analysis, except where the requirements are not applicable.
FLPR-C2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FLPR-B9) associated with the Internal Flood Plant Response model development.
FLPR-C3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details. See FLPR-B10 See Note FL-N-4

4.3.8.6 Objectives and Technical Requirements for Internal Flood Human Reliability Analysis (FLHR)

The objectives of the Internal Flood Human Reliability Analysis are to identify the HFEs, quantify the HEPs for those HFEs to which they apply, and document the Internal Flood Human Reliability Analysis in such a way that

(a) existing HFEs (e.g., from the internal events PRA) are modified to include internal-flood specific performance shaping factors;
(b) flood area-specific and flood scenario-specific human actions are included;
(c) internal flood procedures and direct operator actions taken to maintain acceptable plant configurations and to achieve safe shutdown are included;
(d) HEPs are quantified;
(e) the Internal Flood Human Reliability Analysis is documented to provide traceability of the analysis.

Table 4.3.8.6-1 High Level Requirements for Internal Flood Human Reliability Analysis

Designator	Requirement
HLR-FLHR-A	The Internal Flood PRA shall identify human actions relevant to the event sequences in the Internal Flood Plant Response model.
HLR-FLHR-B	The Internal Flood PRA shall include human failure events in the Internal Flood Plant Response model.
HLR-FLHR-C	The Internal Flood PRA shall quantify HEPs accounting for the plant- or design-specific and scenario-specific influences on human performance, particularly including the effects of internal floods, and addressing potential dependencies.
HLR-FLHR-D	The Internal Flood PRA shall include recovery actions only if it has been demonstrated that the actions are plausible and feasible for those flood scenarios to which they apply.
HLR-FLHR-E	Documentation of the Internal Flood Human Reliability Analysis shall provide traceability of the analysis.

Table 4.3.8.6-2 Supporting Requirements for HLR-FLHR-A

The Internal Flood PRA shall identify human actions relevant to the event sequences in the FLP model. (HLR-FLHR-A)

Index No. FLHR-A	Capability Category I	Capability Category II
FLHR-A1	REVIEW all post-initiator HFEs in the internal events PRA model to determine whether each operator action remains relevant and feasible in the context of the internal flood PRA consistent with the plant response model for the internal flood events and their associated scenarios per the plant response model requirements in this technical area. In determining the applicability of operator actions from the internal events PRA, SATISFY the CC-I SRs of HLR-HR-E for Human Reliability Analysis except where the requirements are not applicable. See Note FL-N-27	REVIEW all post-initiator HFEs in the internal events PRA model to determine whether each operator action remains relevant and feasible in the context of the internal flood PRA consistent with the plant response model for the internal flood events and their associated scenarios per the plant response model requirements in this technical area. In determining the applicability of operator actions from the internal events PRA, SATISFY the CC-II SRs of HLR-HR-E for Human Reliability Analysis except where the requirements are not applicable. See Note FL-N-27
FLHR-A2	For Internal Flood Scenarios Development, IDENTIFY any new internal flood operator actions stated in the plant procedures in a manner consistent with the plant response model for the internal flood events and their associated scenarios per the plant response model requirements in this Section. For any new operator actions identified, SATISFY the CC-I SRs of HLR-HR-E for Human Reliability Analysis except where the requirements are not applicable. See Note FL-N-27	For Internal Flood Scenarios Development, IDENTIFY any new internal flood operator actions stated in the plant procedures in a manner consistent with the plant response model for the internal flood events and their associated scenarios per the plant response model requirements in this Section. For any new operator actions identified, SATISFY the Capability Category II SRs of HLR-HR-E for Human Reliability Analysis except where the requirements are not applicable. For the internal flood events and their associated scenarios per the plant response model, IDENTIFY any undesired operator actions (e.g., terminating a mitigation action) that could result from failures of indicators and annunciators caused by internal flood-induced failure mechanisms. See Note FL-N-27

Table 4.3.8.6-3 Supporting Requirements for HLR-FLHR-B

The Internal Flood PRA shall include human failure events in the Internal Flood Plant Response model. (HLR-FLHR-B)

Index No. FLHR-B	Capability Category I	Capability Category II
FLHR-B1	INCLUDE and, if necessary, MODIFY HFEs corresponding to the operator actions identified per Requirement FLHR-A1 in the Internal Flood Plant Response model in a manner consistent with the modeling, such that the HFEs represent the impact of human failures at the function, system, train, or component level as appropriate.	
FLHR-B2	INCLUDE new internal flood-related HFEs corresponding to the actions identified per Requirement FLHR-A2 and SATISFY the CC-I SRs of HLR-HR-F for Human Reliability Analysis except where the requirements are not applicable.	INCLUDE new internal flood-related HFEs corresponding to the actions identified per Requirement FLHR-A2 and SATISFY the CC-II SRs of HLR-HR-F for Human Reliability Analysis except where the requirements are not applicable.
FLHR-B3	COMPLETE the definition of the HFEs identified in Requirements FLHR-B1 and FLHR-B2 including the relevant internal flood-related context presented by the internal flood events in the PRA at a high level (e.g., sufficient to provide the context needed for a screening Human Reliability Analysis). For the definitions of HFEs, SATISFY the CC-I SRs of HLR-HR-F except where the requirements are not applicable. See Note FL-N-28	COMPLETE the definition of the HFEs identified in Requirements FLHR-B1 and FLHR-B2 including the relevant internal flood-related context presented by the internal flood events in the PRA. For the definitions of HFEs, SATISFY the CC-II SRs of HLR-HR-F except where the requirements are not applicable. See Note FL-N-28

Table 4.3.8.6-4 Supporting Requirements for HLR-FLHR-C

The Internal Flood PRA shall quantify HEPs accounting for the plant- or design-specific and scenario-specific influences on human performance, particularly including the effects of internal floods, and addressing potential dependencies. (HLR-FLHR-C)

Index No. FLHR-C	Capability Category I	Capability Category II
FLHR-C1	CALCULATE the HEPs for all HFEs by addressing relevant internal flood-related effects using conservative estimates (e.g., screening values). For the calculations of HEPs, SATISFY the CC-I SRs of HLR-HR-G for Human Reliability Analysis except where the requirements are not applicable. See Note FL-N-29	CALCULATE the HEPs for all HFEs by addressing relevant internal flood-related effects using detailed analyses for the HFEs that are risk-significant contributors. For the calculations of HEPs, SATISFY the CC-II SRs of HLR-HR-G for Human Reliability Analysis except where the requirements are not applicable. See Note FL-N-29

Table 4.3.8.6-5 Supporting Requirements for HLR-FLHR-D

The Internal Flood PRA shall include recovery actions only if it has been demonstrated that the actions are plausible and feasible for those flood scenarios to which they apply. (HLR-FLHR-D)

Index No. FLHR-D	Capability Category I	Capability Category II
FLHR-D1	IDENTIFY internal flood-specific operator recovery actions and meet Requirement HR-H1 for Human Reliability Analysis. QUANTIFY the corresponding HEP values, including relevant internal flood-related effects and any effects that may preclude a recovery action or alter the manner in which it is accomplished and meet Requirements HR-H1, HR-H2, and HR-H4 for Human Reliability Analysis except where the requirements are not applicable.	
FLHR-D2	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the Internal Flood Human Reliability Analysis in a manner that supports Requirement FLESQ-E1.	
FLHR-D3	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Internal Flood Human Reliability Analysis. See Note FL-N-4, FL-N-7	

Table 4.3.8.6-6 Supporting Requirements for HLR-FLHR-E

Documentation of the internal flood Human Reliability Analysis shall provide traceability of the analysis. (HLR-FLHR-E)

Index No. FLHR-E	Capability Category I	Capability Category II
FLHR-E1	DOCUMENT the process used in the internal flood Human Reliability Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the identification of HFEs, including those carried over from the internal events PRA, new internal flood-specific human actions, recovery actions, and undesired operator actions; *(b)* those internal flood-related influences that affect the methods, processes, or assumptions used; *(c)* flood area-specific and internal flood scenario-specific performance shaping factors for the HFEs identified; *(d)* procedural guidance, training, and plant practice for the operator actions evaluated; *(e)* quantification of HEPs; *(f)* impact of plant operating state changes within the scope of the PRA on Internal Flood Human Reliability Analysis; and *(g)* Meet the documentation SRs of HLR-HR-I for Human Reliability Analysis except where the requirements are not applicable.	
FLHR-E2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirements FLHR-D2) associated with the Internal Flood Human Reliability Analysis.	
FLHR-E3	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Internal Flood Human Reliability Analysis. See FLHR-D3 See Note FL-N-4	

4.3.8.7 Objectives and Technical Requirements for Internal Flood Event Sequence Quantification (FLESQ)

The objectives of Internal Flood Event Sequence Quantification are to quantify the internal flood-induced event sequence frequencies and document the analysis in a way such that

(a) the frequency of flood-induced event sequences and event sequence families are quantified;

(b) flood-induced equipment failures and failures due to independent causes are included in the Event Sequence Quantification;

(c) all identified dependencies (including operator actions) are addressed appropriately as determined by model quantification;

(d) flood-induced event sequences and basic events are identified, evaluated, and understood in the context of the plant design, operation, and maintenance;

(e) analysis limitations and uncertainties are understood; and

(f) the Internal Flood Event Sequence Quantification is documented to provide traceability of the analysis.

Table 4.3.8.7-1 High Level Requirements for Internal Flood Event Sequence Quantification

Designator	Requirement
HLR-FLESQ-A	The frequency of each modeled flood-induced event sequence and event sequence family shall be quantified.
HLR-FLESQ-B	The Internal Flood Event Sequence Quantification shall use appropriate models and codes and a truncation level sufficiently low to show convergence and shall address method-specific limitations and features.
HLR-FLESQ-C	Model quantification shall determine that all identified dependencies (including operator actions) are addressed appropriately.
HLR-FLESQ-D	The Internal Flood Event Sequence Quantification shall be reviewed for correctness, completeness, and consistency. The risk-significant contributors to internal flood-induced event sequences and event sequence families shall be identified, such as plant operating states, flood-induced initiating events, flood-induced event sequences, basic events (equipment unavailability and human failure events), flood areas, flood sources, event phenomena, radionuclide transport barrier failure modes, and other basic events that may influence the mechanistic source terms and radiological consequences. The results shall be traceable to the inputs and assumptions made in the Internal Flood PRA.
HLR-FLESQ-E	Uncertainties in the Internal Flood PRA results shall be characterized. Sources of model uncertainty and related assumptions shall be identified and their potential impact on the results understood.
HLR-FLESQ-F	Documentation of the Internal Flood Event Sequence Quantification shall provide traceability of the analysis and provide interpretation of the risk profile for the plant.

Table 4.3.8.7-2 Supporting Requirements for HLR-FLESQ-A

The frequency of each modeled flood-induced event sequence and event sequence family shall be quantified. (HLR-FLESQ-A)

Index No. FLESQ-A	Capability Category I	Capability Category II
FLESQ-A1	INCLUDE, in the quantification, event sequences comprising failures caused by the flood and those due to independent causes, including equipment failures, unavailability due to maintenance, common cause failures (CCFs), and other credible causes that may reduce the plant capabilities to mitigate the flood-induced initiating event.	
FLESQ-A2	INCLUDE, in the quantification, both the direct effects of the flood (e.g., loss of cooling from a service water train due to an associated pipe rupture) and spatial effects such as submergence, jet impingement, spray, harsh environment, and pipe whip, as applicable.	
FLESQ-A3	If additional analysis of data is required to support the quantification of flood-induced event sequences, PERFORM the necessary Data Analysis and SATISFY the CC-I SRs of HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D for Data Analysis except where the requirements are not applicable.	If additional analysis of data is required to support the quantification of flood-induced event sequences, PERFORM the necessary Data Analysis and SATISFY the CC-II SRs of HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D for Data Analysis except where the requirements are not applicable.
FLESQ-A4	CALCULATE the internal flood-induced event sequence family frequency, on a plant-year basis, and meet the CC-I SRs of HLR-ESQ-A for quantification except where the requirements are not applicable. INCLUDE the scenario-specific quantification factors (e.g., the HEPs obtained per Internal Flood Human Reliability Analysis).	CALCULATE the internal flood-induced event sequence family frequency, on a plant-year basis, and meet the CC-II SRs of HLR-ESQ-A for quantification except where the requirements are not applicable. INCLUDE the scenario-specific quantification factors (e.g., the HEPs obtained per Internal Flood Human Reliability Analysis).
FLESQ-A5	RETAIN internal flood scenarios in the internal flood PRA model unless it can be concluded that SCR-1 or SCR-2 in Table 1.10-1 is satisfied for the affected flood-induced event sequence families.	
FLESQ-A6	COLLECT inputs to the following analyses, which support quantifications of flood-induced event sequences, from plant or design information sources, as applicable, or via investigation(s): *(a)* engineering analyses; *(b)* human reliability analyses; *(c)* spray or other applicable impact assessments; and *(d)* screening decisions. CONFIRM the accuracy of information collected by conducting investigations(s). See Note FL-N-5	
FLESQ-A7	IDENTIFY the sources of model uncertainty and related assumptions associated with the internal flood event sequences and quantification in a manner that supports Requirement FLESQ-E1.	
FLESQ-A8	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details the Internal Flood Event Sequence Quantification. See Note FL-N-4, FL-N-7	

Table 4.3.8.7-3 Supporting Requirements for HLR-FLESQ-B

The Internal Flood Event Sequence Quantification shall use appropriate models and codes and a truncation level sufficiently low to show convergence and shall address method-specific limitations and features. (HLR-FLESQ-B)

Index No. FLESQ-B	Capability Category I	Capability Category II
FLESQ-B1	For the quantification of internal flood-induced event sequence families, SATISFY SRs of HLR-ESQ-B for quantification except where the requirements are not applicable.	

Table 4.3.8.7-4 Supporting Requirements for HLR-FLESQ-C

Model quantification shall determine that all identified dependencies (including operator actions) are addressed appropriately. (HLR-FLESQ-C)

Index No. FLESQ-C	Capability Category I	Capability Category II
FLESQ-C1	INCLUDE dependencies during the internal flood PRA model quantification and SATISFY SRs of HLR-ESQ-C for quantification except where the requirements are not applicable.	

Table 4.3.8.7-5 Supporting Requirements for HLR-FLESQ-D

The Internal Flood Event Sequence Quantification shall be reviewed for correctness, completeness, and consistency. The risk-significant contributors to internal flood-induced event sequences and event sequence families shall be identified, such as plant operating states, flood-induced initiating events, flood-induced event sequences, basic events (equipment unavailability and human failure events), flood areas, flood sources, event phenomena, radionuclide transport barrier failure modes, and other basic events that may influence the mechanistic source terms and radiological consequences; the results shall be traceable to the inputs and assumptions made in the Internal Flood PRA. (HLR-FLESQ-D)

Index No. FLESQ-D	Capability Category I	Capability Category II
FLESQ-D1	IDENTIFY risk-significant contributors and SATISFY the CC-I SRs of HLR-ESQ-D, except where the requirements are not applicable, with the following clarifications: *(a)* Requirements ESQ-D6 and ESQ-D7 are to be met, including identification of which internal flood scenarios and which flood areas are risk-significant contributors. *(b)* Requirement ESQ-D7 is to be met, recognizing that "component" in internal events PRA model is generally equivalent to "equipment" in internal flood PRA. *(c)* Requirement ESQ-D4 for comparison to similar plants is not applicable.	IDENTIFY risk-significant contributors and SATISFY the CC-II SRs of HLR-ESQ-D except where the requirements are not applicable, with the following clarifications: *(a)* Requirements ESQ-D6 and ESQ-D7 are to be met, including identification of which internal flood scenarios and which flood areas are risk-significant contributors. *(b)* Requirement ESQ-D7 is to be met, recognizing that "component" in internal events PRA model is generally equivalent to "equipment" in internal flood PRA. *(c)* Requirement ESQ-D4 for comparison to similar plants is not applicable.

Table 4.3.8.7-6 Supporting Requirements for HLR-FLESQ-E

Uncertainties in the Internal Flood PRA results shall be characterized. Sources of model uncertainty and related assumptions shall be identified and their potential impact on the results understood. (HLR-FLESQ-E)

Index No. FLESQ-E	Capability Category I	Capability Category II
FLESQ-E1	SATISFY Requirement ESQ-E1 for each internal flood technical subelement (see FLPP-B7, FLSO-A8, FLSN-A20, FLEV-B6, FLPR-B9, FLHR-D2, and FLESQ-A7).	
FLESQ-E2	PERFORM an uncertainty analysis for the quantification of flood-induced event sequence families and SATISFY the CC-I Requirement ESQ-E2. See Note FL-N-30	PERFORM an uncertainty analysis for the quantification of flood-induced event sequence families and SATISFY the CC-II Requirement ESQ-E2. See Note FL-N-30

Table 4.3.8.7-7 Supporting Requirements for HLR-FLESQ-F

Documentation of the Internal Flood Event Sequence Quantification shall provide traceability of the analysis and provide interpretation of the risk profile for the plant. (HLR-FLESQ-F)

Index No. FLESQ-F	Capability Category I	Capability Category II
FLESQ-F1	DOCUMENT the process used in the internal flooding risk quantification analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the Internal Flood Event Sequence Quantification process, including any screening performed; *(b)* the results of the Internal Flood Event Sequence Quantification; *(c)* importance measures; *(d)* uncertainty distribution from propagation of parametric uncertainties; *(e)* impact of plant operating state changes on uncertainty quantification. SATISFY the CC-I SRs of HLR-ESQ-F and HLR-DA-E for Data Analysis except where the requirements are not applicable, with the following clarifications: *(a)* Requirements ESQ-F2 and ESQ-F3 are to be met, including identification of which internal flood scenarios and which flood areas are risk-significant contributors. *(b)* Requirement DA-E2 is to be met consistently with Requirement FLESQ-A3. *(c)* Requirement ESQ-F4 is to be met consistently with Requirement FLESQ-E2.	DOCUMENT the process used in the internal flooding risk quantification analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the Internal Flood Event Sequence Quantification process, including any screening performed; *(b)* the results of the Internal Flood Event Sequence Quantification; *(c)* importance measures; *(d)* uncertainty distribution from propagation of parametric uncertainties; *(e)* impact of plant operating state changes on uncertainty quantification. SATISFY the CC-II SRs of HLR-ESQ-F and HLR-DA-E for Data Analysis except where the requirements are not applicable, with the following clarifications: *(a)* Requirements ESQ-F2 and ESQ-F3 are to be met, including identification of which internal flood scenarios and which flood areas are risk-significant contributors. *(b)* Requirement DA-E2 is to be met consistently with Requirement FLESQ-A3. *(c)* Requirement ESQ-F4 is to be met consistently with Requirement FLESQ-E2.
FLESQ-F2	DOCUMENT the risk-significant contributors (e.g., initiating events, event sequences, cutsets, basic events, flood areas, flood sources, operator actions) to risk-significant internal flood-induced event sequences and event sequence families in the PRA results summary. DESCRIBE risk-significant event sequences. Risk-significant flood-induced event sequences and event sequence families depend on both frequency and consequence and are determined within the scope of the technical requirements of the Risk Integration element.	
FLESQ-F3	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FLESQ-A7) associated with the internal flood-induced event sequences and quantification.	
FLESQ-F4	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the internal flood-induced event sequences and quantification. See FLESQ-A8 See Note FL-N-4	
FLESQ-F5	DOCUMENT limitations in the internal flood event sequence quantification process that would impact applications.	

4.3.8.8 Peer Review Requirements for Internal Flood PRA

4.3.8.8.1 Purpose

This Section provides requirements for peer review of the Internal Flood PRA.

4.3.8.8.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the areas of plant partitioning and flood area identification, flood source identification and characterization, flood propagation and flood scenario development, flood event and flood scenario frequency evaluation, flood impact evaluation, flood scenario quantification, flood-induced event sequence modeling, and flood Human Reliability Analysis as applicable to the scope of the review. The team members assigned to internal flood analyses shall have experience specific to these areas and the capability of recognizing plant-specific features of the analyses.

4.3.8.8.3 Review of Internal Flood PRA Subelements to Confirm the Methodology

4.3.8.8.3.1 Internal Flood Plant Partitioning

A review shall be performed on the Internal Flood Plant Partitioning analysis. The portion of the plant partitioning analysis verification typically includes the following:

(a) the overall analysis boundary is appropriate for the internal flood PRA scope;

(b) the criteria used to partition the plant into physical analysis units (flood areas) were defined and appropriate;

(c) the physical analysis units (flood areas) were identified and described;

(d) a selective review by investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable] is recommended to confirm the accuracy of the information obtained from plant information sources to assess spatial information and plant-design features that support the development of flood areas.

4.3.8.8.3.2 Internal Flood Sources Identification and Characterization

A review shall be performed on the internal flood source identification and characterization process. This portion of the process selected for review typically includes the following:

(a) The potential flood sources have included equipment located in flood areas that are connected to fluid systems, internal sources, and external sources that are connected to the flood areas.

(b) The flooding mechanisms have included pressure boundary failure and human-induced events that result in releases in the flood area.

(c) The flood areas screened out do not contain potential flood sources and do not serve as a propagation path to other flood areas. The screening criteria have been uniformly applied and flood areas that are risk-significant contributors are included.

(d) The flood source and corresponding release mechanisms have been appropriately characterized.

(e) A selective review by investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable] is recommended to confirm the accuracy of the information obtained from plant information sources to assess the location of flood sources.

4.3.8.8.3.3 Internal Flood Scenarios Development

A review shall be performed on the Internal Flood Scenarios Development analysis. The portion of the Internal Flood Scenarios Development analysis selected for review typically includes the following:

(a) For a selected set of flood areas and corresponding flood sources, the potential propagation paths have been identified and plant-design features capable of containing or terminating flood propagation are included and appropriately credited.

(b) For a selected set of potential propagation paths, the SSCs along each propagation path represent those that were included in the internal-events PRA model, required to respond to an initiating event, or whose failure would challenge plant operation and are susceptible to flood damage.

(c) The capacity of drains was estimated to determine flood volume and potential impact on PRA-related SSCs.

(d) The susceptibility of SSCs in a selected set of flood areas was determined, and failure of SSCs caused by submergence and spray was considered in the determination process. Flood-induced failure mechanisms other than submergence or spray were assessed.

(e) For a selected set of flood scenarios, the associated flood area and flood source, characteristics of the release, operator actions, and SSCs impacted along the propagation paths were used to develop and define each scenario in a consistent manner.

(f) For a selected set of flood scenarios, the associated calculations included flood source inventory, release rates, propagation pathways, barrier failures, and maximum or critical flood heights for susceptible SSCs in each affected flood area to ensure reasonable characterization of the flood consequence.

(g) The flood areas and flood sources screened out were properly identified, and the bases for screening were applied appropriately.

(h) A selective review by investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable] is recommended to confirm the accuracy of the information obtained from plant information sources to assess the appropriateness of flood scenarios.

4.3.8.8.3.4 Internal Flood Induced Initiating Events

A review shall be performed on the Internal Flood Initiating Events. The portion of the flood-induced event analysis selected for review typically includes the following:

(a) For a selected set of flood scenarios, the corresponding initiating event group for internal events and failures of SSCs caused by a flood have been identified. New initiating event groups have been developed for flood scenarios that had no corresponding plant-initiating event group for internal events.

(b) The grouping of flood scenarios was performed consistently, and the bases for the groupings included plant response, success criteria, timing, equipment, and operator performance.

(c) For selected scenarios, the flood initiating event frequencies were estimated by combining plant-specific and generic information. The frequencies for human-induced floods were also estimated.

(d) The flood scenario groups screened out were properly identified, and the bases for screening were applied appropriately.

4.3.8.8.3.5 Internal Flood Plant Response Model

A review shall be performed on the Internal Flood Plant Response model. The Internal Flood Plant Response model verification typically includes the following:

(a) the plant response model is capable of determining flood-induced radionuclide release and of identifying the risk-significant contributors to the flood-induced risk;

(b) the equipment (e.g., SSCs, instrumentation, barriers) is properly modeled and account for the appropriate flood related failure impacts;

(c) the modeled equipment and HFEs represent the as-built, as-operated plant considering the reactor type, design vintage, and specific design;

(d) the HFEs are properly modeled, including both non–flood-specific and flood-related actions;

(e) findings associated with the internal events analysis have been dispositioned such that they do not adversely impact the internal flood PRA.

4.3.8.8.3.6 Internal Flood Human Reliability Analysis

A review shall be performed on the Internal Flood Human Reliability Analysis. The portion of the Internal Flood Human Reliability Analysis selected for review typically includes the following:

(a) for a selected set of flood-induced scenarios, the corresponding HFEs were identified to determine their applicability;

(b) the reliability of internal flood procedure and direct operator actions were included in the HEP quantification;

(c) performance issues were included in the HEP quantifications to which they apply.

(d) the Internal Flood Human Reliability Analysis was performed consistently with the applicable requirements of the Internal Flood Human Reliability Analysis technical element, and all scenario-specific impacts on performance shaping factors were included;

(e) a selected review of investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable] was conducted to confirm the feasibility of operator action to mitigate the internal flood.

4.3.8.8.3.7 Internal Flood Event Sequence Quantification

A review shall be performed on the Internal Flood Event Sequence Quantification. The portion of the internal flood event sequence and quantification analysis selected for review typically includes the following:

(a) for a selected set of flood-induced scenarios, the corresponding sequences for the plant-initiating event are applicable;

(b) the flood-induced scenarios screened out at this level were identified, and the screening was performed appropriately;

(c) the flood event sequences were quantified in accordance with the applicable requirements in Event Sequence Quantification and the combined effects of flood-induced failures of SSCs were properly analyzed;

(d) for selected flood event sequences, the contribution to event sequence frequency was evaluated correctly;

(e) a selective review by investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable] is recommended to confirm the accuracy of the information obtained from plant information sources to assess the appropriateness of Human Reliability Analysis, spray or other impact assessment, and engineering analyses on the quantification results.

4.3.8.9 References for Internal Flood PRA

The following is a list of publications referenced in this Standard. Unless otherwise specified, the latest edition shall apply.

[FL-1] EPRI 1019194, "Guidelines for Performance of Internal Flooding Probabilistic Risk Assessment," Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304, 2009

[FL-2] EPRI reports on "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments," Publisher: Electric Power Research Institute (EPRI), 3420 Hillview Avenue, Palo Alto, CA 94304

[FL-3] Nuclear Power Operations' EPIX (Equipment Performance and Information Exchange System)

NONMANDATORY APPENDIX FL: NOTES AND EXPLANATORY MATERIAL FOR INTERNAL FLOOD PRA

FL.1 NOTES ASSOCIATED WITH INTERNAL FLOOD PRA

Table FL-1 Notes Supporting Internal Flood PRA Requirements

Number	Notes
FL-N-1	The collection of data for PRAs with multiple plant operating states includes verification of any applicable temporary alignments for the specific plant evolution or evolutions being modeled in the PRA model. Note that the arrangement of flood barriers, flood propagation paths, and equipment susceptibilities to flooding may be different in different plant operating states within the scope of the PRA. For example, this includes opened/impaired hazard doors, opened covers on drains, and additional sources of floods. For outage work activities with potential for temporary impairment of flood doors/barriers and potential for maintenance-induced floods, risk management actions are required and may include limiting the allowed impairment time of flood barriers and using compensatory measures and contingency plans. See FLPP-B1, FLSN-A1, FLSN-A10, FLSN-A13
FL-N-2	This SR is not applicable to PRAs performed for a single reactor plant. See FLPP-B2, FLSO-A3, FLEV-A3
FL-N-3	This SR is likely not applicable to PRAs performed during the pre-operational stage. See FLPP-B3
FL-N-4	This SR is not applicable to operating plants. See FLPP-B4, FLPP-B8, FLPP-C3, FLSO-A9, FLSO-B3, FLSN-A21, FLSN-B3, FLEV-B7, FLEV-C3, FLPR-B10, FLPR-C3, FLHR-D3, FLHR-E3, FLESQ-A8, FLESQ-F4
FL-N-5	Examples of investigations include, but are not limited to, actives such as tabletop review(s), physical/computerized walkdown(s) of the plant, and/or interview(s) with knowledgeable personnel. These investigation(s) may need to be supported by a review of plant documents, especially in the context of external hazard evaluations. This SR does not imply that each of these activities need to be performed to meet the SR. Further, if appropriate, one investigation can meet the intent of multiple investigation SRs if the investigation is scoped and structured appropriately. See FLPP-B6, FLSO-A7, FLSN-A19, FLESQ-A6
FL-N-6	For operating plants, the basis for judging the adequacy of the level of detail, fidelity, and realism of the PRA is the available set of characteristics of the as-built and as-operated plant. See FLPP-B6
FL-N-7	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See FLPP-B8, FLSO-A9, FLSN-A21, FLEV-B7, FLPR-B10, FLHR-D3, FLESQ-A8
FL-N-8	An example of a flood area partitioning element is a room with enclosed walls and door, a portion of an area separated from other parts of the area with a curb. See FLPP-C1
FL-N-9	Examples of flood sources include the following: *(a)* equipment flood sources: circulating water system, service water system, component cooling water system, fire protection system, feedwater system, condensate and steam systems, reactor coolant system, and other high-energy lines; *(b)* plant internal flood sources: tanks or pools; *(c)* external flood sources: reservoirs or rivers. See FLSO-A1

Table FL-1 Notes Supporting Internal Flood PRA Requirements (Cont'd)

Number	Notes
FL-N-10	Sources of flooding are typically expected to be water, and the requirements are generally written in terms of sources of water, but other fluid sources should also be considered as part of this standard. Note that flood sources may be different in different plant operating states within the scope of the PRA. See FLSO-A1
FL-N-11	Maintenance-induced flooding may have increased potential during plant operating states associated with low power and shutdown (LPSD) evolutions. See FLSO-A5, FLEV-B2
FL-N-12	Examples of flood types are leak, rupture, or spray. See FLSO-A6
FL-N-13	Careful consideration of the dependence of flood mitigation features on each modeled plant operating state and the possibility that such features may be compromised during the specific plant operating state is necessary. See FLSN-A2, FLSN-A8
FL-N-14	Automatic responses during plant operating states in LPSD evolutions are likely to differ from full power; examples of flood scenarios originating when no one is watching (filling is going on, and workers are on break) are apparent in flood data. See FLSN-A3
FL-N-15	Examples of mitigating features are spray shielding, equipment enclosure ratings for flood, or spray proofing. See FLSN-A5
FL-N-16	Examples of flood-induced mechanisms not explicitly addressed are jet impingement or pipe whip. See FLSN-A6
FL-N-17	Examples of values for flood area design features are flood area dimensions, floor opening dimensions, wall opening dimensions, floor and door gap dimensions, drain sizes, free volume not occupied by SSCs, and SSC critical flood heights. See FLSN-A9
FL-N-18	Flood scenarios may be different for each plant operating state. See FLSN-A13
FL-N-19	For PRAs performed during the pre-operational stage, the flood scenarios should be limited to a level of detail that is consistent with the level of detail of the design information available to support the identification of risk-significant event sequences and event sequence families. See FLSN-A13
FL-N-20	This SR is not applicable to PRAs performed for a single source of radioactive material. See FLSN-A14, FLSN-A15, FLEV-A4
FL-N-21	An example of chemically or physically incompatible fluid impacts is sodium, which reacts exothermically with water. If a holding tank of activated sodium leaks during an internal flood scenario, the impact of released energy and hydrogen gas generation from this reaction should be assessed. See FLSN-A16
FL-N-22	The use and extent of screening out of flood areas and sources are optional. To facilitate an efficient qualitative screening process, conservative representations of the flood impact may be used for screening purposes. Examples of conservative representations include the use of bounding assumptions on flood rate, flood volume, barrier effectiveness, mitigation, and SSC susceptibility to flood-induced failure mechanisms. The qualitative screening criteria for flood sources and areas in Requirements FLSN-A17 and FLSN-A18 may be used in conjunction with conservative representations of scenarios. For areas and sources not screened out under the requirements of Requirements FLSN-A17 or FLSN-A18, flood scenarios will need to be defined. The requirements in Internal Flood Initiating Events and Internal Flood Event Sequence Quantification that apply to flood scenarios require a realistic representation and enumeration of flood scenarios unless otherwise noted. See FLSN-A17, FLSN-A18

Table FL-1 Notes Supporting Internal Flood PRA Requirements (Cont'd)

Number	Notes
FL-N-23	The wording of this requirement recognizes that flood sources and flood areas may be screened out prior to the task of enumerating all relevant flood scenarios for each source and area to facilitate an efficient screening process. For the purpose of defining the limiting flood for Requirement FLSN-A13 and the most challenging flood for Requirement FLSN-A14, it is necessary to ensure that all the parameters relevant to flood scenario definition are considered and bounded when defining these floods. See FLSN-A18
FL-N-24	Databases such as Institute of Nuclear Power Operations Equipment Performance and Information Exchange System (EPIX), lessons learned from industry LPSD evolutions, and lessons learned from self-assessment of previous LPSD evolutions are good sources for identifications of flood-induced initiating events and their frequencies. See FLEV-A2, FLEV-B3
FL-N-25	Generic examples of piping system failure rates for use in estimating flood-induced initiating event frequencies may be found in the most recent Electric Power Research Institute (EPRI) report on "Pipe Rupture Frequencies for Internal Flooding Probabilistic Risk Assessments." [FL-2] See FLEV-B3
FL-N-26	The internal events PRA model may or may not include event sequences involving two or more reactors or radionuclide sources. However, if an internal flood and its propagation paths impacts two or more reactors or sources, such event sequences may be introduced for treatment of this hazard group. See FLPR-B1
FL-N-27	This SR has the following clarifications when backreferencing to HLR-HR-E: *(a)* where Requirement HR-E1 specifies "in the context of the scenarios," include the effects resulting from the internal flood events; and *(b)* where Requirement HR-E1 specifies procedures, they are to include procedures for responding to conditions that can be caused by internal floods. See FLHR-A1, FLHR-A2
FL-N-28	The HLR-HR-F backreference should include considerations of flood indication availability and expected time available for human response actions to be performed for the most challenging flood for the flood sources being addressed. See FLHR-B3
FL-N-29	This SR has the following clarification: Attention is to be given to how the internal flood situation alters any previous assessments in non-internal flood analyses as to the influencing factors and the timing considerations covered in Requirements HR-G2, HR-G3, HR-G4, and HR-G7 for Human Reliability Analysis. See FLHR-C1
FL-N-30	In general, flood-induced event sequences will comprise a combination of initiating events and basic events associated with the following: *(a)* flood-induced initiating events; *(b)* portions of the event sequences derived from the internal events PRA model (i.e., basic events that are independent of the flood scenarios but otherwise contribute to the event sequence). Hence, the sources of model uncertainty that impact quantification include a combination of uncertainties associated with the flood scenarios and flood-induced initiating events as well as those that are carried over from the internal events PRA model. These requirements (including Requirement ESQ-E1) include all sources of model uncertainty that impact the flood-induced Event Sequence Analysis. See FLESQ-E2

FL.2 EXPLANATORY CONTENT ASSOCIATED WITH INTERNAL FLOOD PRA

The scope of the flooding events includes all flood scenarios originating within the plant boundary. It does not include floods resulting from external events (e.g., weather, or off-site events such as upstream dam rupture).

The overall objective of the internal flood PRA is to ensure that the impact of internal flood as the cause of either an event or a system failure is evaluated in such a way that

(a) the flood sources within the plant that could flood plant locations or create adverse conditions (e.g., submersion, spray, elevated temperature, humidity, pressure, pipe whip, jet impingement) that could damage mitigative plant equipment are identified;

(b) the flood-induced event sequences that contribute to the event sequence frequency are identified and quantified.

A separate set of technical elements and associated requirements is provided for this hazard group in this Standard because there are many different sources of floods throughout the plant, with different potential impact on SSCs. Thus, there is the potential for a relatively large number of individual flood scenarios and flood-induced event sequences with unique spatial dependencies. Although it is optional, some degree of screening out of flood-induced scenarios and event sequences is typically employed in analyzing risk from internal floods, so that, although the HLRs and SRs are written in a discrete manner, the requirements are not necessarily presented in sequential order of application and, in some cases, must be considered jointly, so that screening out is performed appropriately. Thus, to determine the degree to which a particular SR is to be met, it is necessary to consider the degree to which other related requirements (some of which may be under other HLRs) are being addressed. Screening out is typically employed at the flood area, flood source, or flood scenario level with the understanding that screening out of areas and sources includes the relevant flood scenarios associated with the area or source.

An internal flood PRA need not be performed at a uniform level of detail. The analyses performed to support the screened-out flood areas may be performed at a less rigorous completeness level than analyses performed for flood areas, flood sources, and/or flood scenarios that are not screened out and hence require further analysis. An iterative process is also common in internal flood PRAs. Those flood areas that represent the higher risk contributors may be analyzed repeatedly, each time incorporating additional detail for specific aspects of the analysis (e.g., flood source and propagation modeling, credit for drains or mitigation, refinements to the Internal Flood Plant Response model, and the Human Reliability Analysis). At any stage, the additional detail may allow for the screening out of a flood area. It is intended that this Standard allows for analysis flexibility in this regard. As such, the level of detail and resolution for lower risk and/or screened-out flood areas may be lower than for higher risk flood areas, which are not screened out, without affecting the capability of the internal flood PRA to identify flood-induced event sequences that are risk-significant contributors. For example, a service building containing numerous flood sources may be analyzed as a single flood area and analyzed for screening purposes. If the building can be screened out (e.g., it contains no equipment modeled in the other portions of the PRA and there are no propagation paths to other buildings), then the overall categorization of the internal flood PRA is unaffected. Similarly, the requirements for developing specific internal flood scenarios, detailed Human Reliability Analysis, etc., are not needed for screened-out flood areas and may not be needed for lower-risk flood areas that are not screened out as long as the overall validity of the final results is unaffected.

Example approaches to performing each of the flood technical subelements of an internal flood PRA may be found in EPRI 1019194 [FL-1].

4.3.9 Internal Fire PRA (F)

This Section presents the technical requirements associated with Internal Fire PRA. The technical requirements for the internal fire hazard group are limited to "at-power" plant operating states as there was insufficient experience in performing internal fire PRAs during low power and shutdown (LPSD) plant operating states to justify the inclusion of such requirements. This exclusion is consistent with supporting LWR PRA standards.

The requirements in this Section are divided into the following technical subelements:

(a) Internal Fire Plant Boundary Definition and Partitioning (FPP);
(b) Internal Fire Equipment Selection (FES);
(c) Internal Fire Cable Selection and Location (FCS);
(d) Internal Fire Qualitative Screening (FQLS);
(e) Internal Fire Plant Response Model (FPRM);
(f) Internal Fire Scenario Selection and Analysis (FSS);
(g) Internal Fire Ignition Frequency (FIGN);
(h) Internal Fire Circuit Failure Analysis (FCF);
(i) Internal Fire Human Reliability Analysis (FHR);
(j) Internal Fire Event Sequence Quantification (FESQ).

4.3.9.1 Objectives and Technical Requirements for Internal Fire Plant Boundary Definition and Partitioning (FPP)

The objectives of the Internal Fire Plant Boundary Definition and Partitioning ensure that

(a) the global analysis boundary of the internal fire PRA, that is, to define the physical extent of the plant to be encompassed by the internal fire analysis;

(b) the physical analysis units (PAU) upon which the analysis will be based; and

(c) the Internal Fire Plant Boundary Definition and Partitioning is documented to provide traceability of the analysis.

Table 4.3.9.1-1 High Level Requirements for Internal Fire Plant Boundary Definition and Partitioning

Designator	Requirement
HLR-FPP-A	The Internal Fire PRA shall define the global analysis boundary to include all plant locations relevant to the plant-wide Internal Fire PRA.
HLR-FPP-B	The Internal Fire PRA shall perform a plant partitioning analysis to identify and define the PAUs to be evaluated in the Internal Fire PRA.
HLR-FPP-C	The documentation of the Internal Fire Plant Boundary Definition and Partitioning shall provide traceability of the work.

Table 4.3.9.1-2 Supporting Requirements for HLR-FPP-A

The Internal Fire PRA shall define the global analysis boundary to include all plant locations relevant to the plant-wide Internal Fire PRA. (HLR-FPP-A)

Index No. FPP-A	Capability Category I	Capability Category II
FPP-A1	INCLUDE within the global analysis boundary all fire areas, fire compartments, or locations within the licensee-controlled area where a fire could adversely affect any equipment or cable item to be included in the Internal Fire Plant Response Model, including those locations where a fire could impact two or more reactors or sources of radioactive material in the scope of the PRA. See Note F-N-1, F-N-2	

Table 4.3.9.1-3 Supporting Requirements for HLR-FPP-B

The Internal Fire PRA shall perform a plant partitioning analysis to identify and define the PAUs to be evaluated in the Internal Fire PRA. (HLR-FPP-B)

Index No. FPP-B	Capability Category I	Capability Category II
FPP-B1	DEFINE a set of internal fire PRA PAUs that represent the physical characteristics of the plant, the nature of the fire hazards presents in each plant location, and the potential extent of fire damage that could reasonably result from fires involving those fire sources. JUSTIFY that the defined PAUs are sufficient and appropriate.	
FPP-B2	If any physical plant feature that lacks a specific fire-endurance rating has been credited as a partitioning element in defining the boundaries of the PAUs (see Requirement FPP-B1), JUSTIFY the judgment that the nonrated partitioning element will substantially contain the damaging effects of fires given the nature of the fire sources present in each PAU, separated by the nonrated partitioning element. See Note F-N-3, F-N-4	
FPP-B3	DO NOT CREDIT raceway fire barriers, thermal wraps, fire-retardant coatings, radiant energy shields, or any other localized cable or equipment protection feature as partitioning elements in defining PAUs.	
FPP-B4	ENSURE that *(a)* collectively, the defined PAUs encompass all locations within the global analysis boundary (see Requirement FPP-A1); *(b)* defined PAUs do not overlap.	
FPP-B5	For operating reactors, COLLECT information on credited barriers that are not maintained as a part of the fire protection program to confirm the conditions and characteristics of credited partitioning elements via a confirmatory investigation. See Note F-N-5, F-N-6	
FPP-B6	JUSTIFY the exclusion of any locations within the licensee-controlled area from the global analysis boundary by demonstrating that they do not satisfy the selection criteria as defined per Requirement FPP-A1.	
FPP-B7	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the PAU identification in a manner that supports Requirement FESQ-E1.	
FPP-B8	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the PAU identification. See Note F-N-7, F-N-8	

Table 4.3.9.1-4 Supporting Requirements for HLR-FPP-C

The documentation of the Internal Fire Plant Boundary Definition and Partitioning analysis shall provide traceability of the work. (HLR-FPP-C)

Index No. FPP-C	Capability Category I	Capability Category II
FPP-C1	DOCUMENT the process used in the internal fire plant partitioning, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of Supporting Requirements (SRs) are satisfied: *(a)* the approach used for developing the internal fire plant partitioning analysis; *(b)* identification of plant documentation used in support of the internal fire plant partitioning; *(c)* the exclusion of any locations within the licensee-controlled area that are not included in the global analysis boundary; *(d)* the general nature and key or unique features of the partitioning elements that define each PAU defined in internal fire plant partitioning; *(e)* the internal fire PRA PAUs; *(f)* the investigation process; *(g)* the results of the internal fire plant partitioning.	
FPP-C2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirements FPP-B7) associated with the internal fire plant partitioning.	
FPP-C3	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details. See FPP-B8 See Note F-N-8	

4.3.9.2 Objectives and Technical Requirements for Internal Fire Equipment Selection (FES)

The objectives of the Internal Fire Equipment Selection technical element are as follows:

(a) identify the fire-induced initiating events that will be included in the Internal Fire Plant Response Model;
(b) identify the plant equipment that will be included in the Internal Fire Plant Response Model;
(c) identify the instruments that will be included in the Internal Fire Plant Response Model;
(d) the Internal Fire Equipment Selection is documented to provide traceability of the analysis.

Table 4.3.9.2-1 High Level Requirements for Internal Fire Equipment Selection

Designator	Requirement
HLR-FES-A	The Internal Fire PRA shall identify the fire-induced initiating events to be evaluated in the Internal Fire Plant Response Model and the equipment whose failure, including spurious operation, would cause each initiating event.
HLR-FES-B	The Internal Fire PRA shall identify equipment whose failure, including spurious operation, would compromise mitigating systems that are included in the Internal Fire PRA.
HLR-FES-C	The Internal Fire PRA shall identify instrumentation whose failure, including spurious operation, would impact the reliability of operator actions associated with that portion of the plant design to be included in the Internal Fire PRA.
HLR-FES-D	The documentation of the Internal Fire Equipment Selection shall provide traceability of the work.

Table 4.3.9.2-2 Supporting Requirements for HLR-FES-A

The Internal Fire PRA shall identify the fire-induced initiating events to be evaluated in the Internal Fire Plant Response Model and the equipment whose failure, including spurious operation, would cause each initiating event. (HLR-FES-A)

Index No. FES-A	Capability Category I	Capability Category II
FES-A1	For each initiating event included in the internal events plant response model, and for each initiating event that was considered but screened out from the internal events plant response model, either INCLUDE the initiating event in the Internal Fire Plant Response Model or JUSTIFY exclusion of the initiating event from the Internal Fire Plant Response Model.	
FES-A2	IDENTIFY the equipment whose fire-induced loss of function failure would cause any of the initiating events that have been included per Requirement FES-A1. See Note F-N-9, F-N-10	
FES-A3	IDENTIFY, by using a structured systematic process that meets the criteria set forth in Requirements FES-A4, FES-A5, and FES-A6, any unique initiating events, and the equipment whose fire-induced failure including spurious operation would cause them, which are not already included per Requirements FES-A1 and FES-A2. See Note F-N-11	
FES-A4	IDENTIFY equipment based on the consideration of cases where any single fire-induced spurious operation of equipment alone would cause an initiating event. See Note F-N-12	

Table 4.3.9.2-2 Supporting Requirements for HLR-FES-A (Cont'd)

The Internal Fire PRA shall identify the fire-induced initiating events to be evaluated in the Internal Fire Plant Response Model and the equipment whose failure, including spurious operation, would cause each initiating event. (HLR-FES-A)

Index No. FES-A	Capability Category I	Capability Category II
FES-A5	IDENTIFY equipment based on the consideration of any single fire-induced spurious operations that, in combination with other fire-induced loss of function failures, would cause an initiating event. See Note F-N-12	IDENTIFY equipment based on the consideration of any single fire-induced spurious operations that, in combination with other fire-induced loss of function failures, would cause an initiating event. IDENTIFY equipment based on the consideration of combinations of two fire-induced spurious operations that alone or in combination with other fire-induced loss of function failures would cause an initiating event and *(a)* affect the portion of the plant design to be credited in response to the initiating event in the internal fire PRA; or *(b)* result in a loss of reactor coolant system integrity. See Note F-N-12
FES-A6	IDENTIFY equipment based on the consideration of up to two fire-induced spurious operations of equipment alone or in combination with other fire-induced loss of function failures that cause an initiating event and containment bypass. See Note F-N-12, F-N-13, F-N-14, F-N-15	IDENTIFY equipment based on the consideration of up to three fire-induced spurious operations of equipment alone or in combination with other fire-induced loss of function failures that cause an initiating event and containment bypass. See Note F-N-12, F-N-13, F-N-14, F-N-15
FES-A7	For any identified equipment from FES-A3, FES-A4, FES-A5, and FES-A6, either INCLUDE the identified equipment in the Internal Fire Plant Response Model or JUSTIFY the exclusion of equipment per the requirements of SCR-2 or SCR-3 in Table 1.10-1. See Note F-N-16	

Table 4.3.9.2-3 Supporting Requirements for HLR-FES-B

The Internal Fire PRA shall identify equipment whose failure, including spurious operation, would compromise mitigating systems that are included in the Internal Fire PRA. (HLR-FES-B)

Index No. FES-B	Capability Category I	Capability Category II
FES-B1	IDENTIFY plant equipment that is both vulnerable to fire-induced failure and whose failure could compromise mitigating systems modeled in the internal fire PRA.	
FES-B2	For every train of equipment that is included in the Internal Fire Plant Response Model, IDENTIFY equipment using a structured systematic process whose fire-induced failures including any single spurious operation will contribute to the failure to meet the success criteria in the internal fire PRA. See Note F-N-17	For every train of equipment that is included in the Internal Fire Plant Response Model, IDENTIFY equipment using a structured systematic process whose fire-induced failures up to and including two spurious operations will contribute to the failure to meet the success criteria in the internal fire PRA. See Note F-N-17
FES-B3	For any identified equipment from FES-B1 and FES-B2, either: INCLUDE the identified equipment in the plant response model or JUSTIFY exclusion of equipment per the requirements of SCR-3 in Table 1.10-1. See Note F-N-16	

Table 4.3.9.2-4 Supporting Requirements for HLR-FES-C

The Internal Fire PRA shall identify instrumentation whose failure, including spurious operation, would impact the reliability of operator actions associated with that portion of the plant design to be included in the Internal Fire PRA. (HLR-FES-C)

Index No. FES-C	Capability Category I	Capability Category II
FES-C1	IDENTIFY instrumentation for which fire-induced failure is relevant in assessing the human failure event (HFEs) that are defined or modified to account for the context of fire scenarios in the Internal Fire PRA, per SRs FHR-B1 and FHR-B2. See Note F-N-18	IDENTIFY instrumentation for which fire-induced failure is relevant in assessing the HFEs that are defined or modified to account for the context of fire scenarios in the Internal Fire PRA, per SRs FHR-B1 and FHR-B2, including the consideration of the following: *(a)* loss of function, loss of signal failures; *(b)* any fire-induced spurious/erroneous indications of a single instrument that would directly lead the operators to take an undesirable action impacting one or more of the safety functions modeled in the Internal Fire PRA. See Note F-N-18
FES-C2	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the internal fire initiating events and equipment selection in a manner that supports Requirement FESQ-E1.	
FES-C3	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the internal fire initiating events and equipment selection. See Note F-N-7, F-N-8	

Table 4.3.9.2-5 Supporting Requirements for HLR-FES-D

The documentation of the Internal Fire Equipment Selection shall provide traceability of the work. (HLR-FES-D)

Index No. FES-D	Capability Category I	Capability Category II
FES-D1	DOCUMENT the process used in the initiating events and equipment selection analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* identification of the equipment associated with determining initiating events in the Internal Fire Plant Response Model for the postulated fires; *(b)* the equipment and their failures modes, including spurious operation or indication to be included in the Internal Fire Plant Response Model. See Note F-N-19	
FES-D2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FES-C2) associated with the Internal Fire Equipment Selection.	
FES-D3	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Internal Fire Equipment Selection. See FES-C3 See Note F-N-8	

4.3.9.3 Objectives and Technical Requirements for Internal Fire Cable Selection and Location (FCS)

The objectives of the Internal Fire Cable Selection and Location ensure that

(a) cables needed to support proper operation of equipment identified in Event Sequence Analysis are identified and assessed for relevance to the Internal Fire Plant Response Model;

(b) the plant location information for the identified cables is sufficient to support the internal fire PRA and its intended applications; and

(c) the Internal Fire Cable Selection and Location is documented to provide traceability of the analysis.

Table 4.3.9.3-1 High Level Requirements for Internal Fire Cable Selection and Location

Designator	Requirement
HLR-FCS-A	The Internal Fire PRA shall identify and locate the plant cables whose failure would adversely affect equipment or functions included in the Internal Fire Plant Response Model, as determined by the equipment selection process per the SRs of HLR-FES-A, HLR-FES-B, and HLR-FES-C.
HLR-FCS-B	The Internal Fire PRA shall perform a review for additional circuits associated with overcurrent protection that are required to support equipment included in the Internal Fire Plant Response Model (i.e., per the SRs of HLR-FCS-A).
HLR-FCS-C	The documentation of the Internal Fire Cable Selection and Location shall provide traceability of the work.

Table 4.3.9.3-2 Supporting Requirements for HLR-FCS-A

The Internal Fire PRA shall identify and locate the plant cables whose failure would adversely affect equipment or functions included in the Internal Fire Plant Response Model, as determined by the equipment selection process per the SRs of HLR-FES-A, HLR-FES-B, and HLR-FES-C. (HLR-FCS-A)

Index No. FCS-A	Capability Category I	Capability Category II
FCS-A1	IDENTIFY, by using a structured and systematic process, cables whose fire-induced failure would adversely affect equipment selected per Internal Fire Equipment Selection and/or functions included in the Internal Fire Plant Response Model with the exception of equipment excluded per Requirement FCS-A2. See Note F-N-20	IDENTIFY, by using a structured and systematic process, cables whose fire-induced failure would adversely affect equipment selected per Internal Fire Equipment Selection and/or functions included in the Internal Fire Plant Response Model with the exception of equipment excluded per Requirement FCS-A2; and for equipment that is a risk-significant contributor, ASSOCIATE cables with equipment failure modes specific to each cable. See Note F-N-20, F-N-21
FCS-A2	IDENTIFY systems and/or equipment selected per Internal Fire Equipment Selection for which cable selection and routing has not been performed; and JUSTIFY that the lack of cable selection and routing does not impact the insights associated with risk-significant contributors. See Note F-N-22, F-N-23	

Table 4.3.9.3-2 Supporting Requirements for HLR-FCS-A (Cont'd)

The Internal Fire PRA shall identify and locate the plant cables whose failure would adversely affect equipment or functions included in the Internal Fire Plant Response Model, as determined by the equipment selection process per the SRs of HLR-FES-A, HLR-FES-B, and HLR-FES-C. (HLR-FCS-A)

Index No. FCS-A	Capability Category I	Capability Category II
FCS-A3	For each PAU, IDENTIFY each cable, including their terminal locations, associated with a function included in the internal fire PRA that passes through the PAU.	For each PAU, IDENTIFY each cable, including their terminal locations, associated with a function (i.e., failure mode or basic event) included in the internal fire PRA that passes through the PAU. For fire scenarios that are risk-significant contributors, IDENTIFY the electrical raceways though which each target cable is routed. If exact routing information is unavailable, meet the requirements of FCS-A4 for assumed routing. See Note F-N-24
FCS-A4	If assumed cable routing is used in the internal fire PRA, SPECIFY the basis for the scope, extent, and basis. See Note F-N-25, F-N-26, F-N-27	

Table 4.3.9.3-3 Supporting Requirements for HLR-FCS-B

The Internal Fire PRA shall perform a review for additional circuits associated with over current protection that are required to support equipment included in the Internal Fire Plant Response Model (i.e., per the SRs of HLR-FCS-A). (HLR-FCS-B)

Index No. FCS-B	Capability Category I	Capability Category II
FCS-B1	ASSESS the adequacy of the electrical overcurrent protective device coordination for distribution buses included in the Internal Fire Plant Response Model.	
FCS-B2	IDENTIFY any additional circuits/cables whose fire-induced failure would challenge power supply availability due to inadequate overcurrent protective device coordination.	
FCS-B3	For PRAs performed during the pre-operational stage, IDENTIFY assumptions regarding overcurrent protection. See Note F-N-8	

Table 4.3.9.3-4 Supporting Requirements for HLR-FCS-C

The documentation of the Internal Fire Cable Selection and Location shall provide traceability of the work. (HLR-FCS-C)

Index No. FCS-C	Capability Category I	Capability Category II
FCS-C1	DOCUMENT the process used in the cable selection and location analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the cable selection and location results such that those results are traceable to plant source documents; *(b)* the assumed cable routing and the basis for concluding that the routing is reasonable if the provision of Requirement FCS-A4 is used; *(c)* the review of the electrical distribution system overcurrent coordination and protection analysis.	
FCS-C2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in the SRs of HLR-FCS-A and HLR-FCS-B) associated with the Internal Fire Cable Selection and Location.	
FCS-C3	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details (as identified in the SRs of HLR-FCS-A and HLR-FCS-B). See Note F-N-8	

4.3.9.4 Objectives and Technical Requirements for Internal Fire Qualitative Screening (FQLS)

The objective of the Internal Fire Qualitative Screening technical element is to

(a) identify physical analysis units whose potential fire risk contribution can be shown to be negligible without quantitative analysis; and

(b) ensure the Internal Fire Qualitative Screening is documented to provide traceability of the analysis.

Table 4.3.9.4-1 High Level Requirements for Internal Fire Qualitative Screening

Designator	Requirement
HLR-FQLS-A	The Internal Fire PRA shall identify those PAUs that screen out as individual risk contributors without quantitative analysis.
HLR-FQLS-B	The documentation of the Internal Fire Qualitative Screening shall provide traceability of the work.

Table 4.3.9.4-2 Supporting Requirements for HLR-FQLS-A

The Internal Fire PRA shall identify those PAUs that screen out as individual risk contributors without quantitative analysis. (HLR-FQLS-A)

Index No. FQLS-A	Capability Category I	Capability Category II
FQLS-A1	RETAIN for quantitative analysis those PAUs that contain equipment or cables required to ensure as-designed circuit operation, or whose failure could cause spurious operation, of any equipment, system, function, or operator action included in the Internal Fire Plant Response Model per SCR-3 in Table 1.10-1. If exact routing information is unavailable, meet the requirements of FCS-A4 for assumed routing. See Note F-N-24, F-N-28	
FQLS-A2	RETAIN for quantitative analysis those PAUs where a fire might require a manual or automatic plant trip or a controlled manual shutdown based on plant Technical Specifications per SCR-3 in Table 1.10-1. If a time limit is established for a required Technical Specifications required shutdown, SPECIFY the basis for the applied time window. See Note F-N-28, F-N-29	
FQLS-A3	APPLY the screening criteria as defined by FQLS-A1 and FQLS-A2 to each PAU defined in the plant partitioning analysis. See Note F-N-28	
FQLS-A4	If additional qualitative screening criteria are applied, SPECIFY the applied criteria and the basis that demonstrates the applied criteria provide reasonable assurance that the screened-out PAUs are negligible contributors to fire risk in a manner consistent, at a minimum, with Requirements FQLS-A1, FQLS-A2, and FQLS-A3. See Note F-N-30	
FQLS-A5	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the Internal Fire Qualitative Screening analysis in a manner that supports Requirement FESQ-E1.	
FQLS-A6	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Internal Fire Qualitative Screening analysis. See Note F-N-7, F-N-8	

Table 4.3.9.4-3 Supporting Requirements for HLR-FQLS-B

The documentation of the Internal Fire Qualitative Screening shall provide traceability of the work. (HLR-FQLS-B)

Index No. FQLS-B	Capability Category I	Capability Category II
FQLS-B1	DOCUMENT the process used in the Internal Fire Qualitative Screening, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the qualitative screening criteria applied; *(b)* the disposition of each PAU defined by the Internal Fire Plant Boundary Definition and Partitioning as either "screened out" or "retained for quantitative analysis"; *(c)* the basis for exclusion of each PAU defined in the Internal Fire Plant Boundary Definition and Partitioning that has been screened out.	
FQLS-B2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FQLS-A5) associated with the Internal Fire Qualitative Screening.	
FQLS-B3	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details. See FQLS-A6 See Note F-N-8	

4.3.9.5 Objectives and Technical Requirements for Internal Fire Plant Response Model (FPRM)

The objective of the Internal Fire Plant Response Model technical element is to

(a) provide the basis for the identification of fire-induced event sequences and event sequence cutsets;
(b) incorporate at-power technical elements into the Internal Fire Plant Response Model; and
(c) document the Internal Fire Plant Response Model to provide traceability of the analysis.

Table 4.3.9.5-1 High Level Requirements for Internal Fire Plant Response Model

Designator	Requirement
HLR-FPRM-A	The Internal Fire PRA shall include the Internal Fire Plant Response Model capable of the SRs of HLR-FESQ-A, HLR-FESQ-B, HLR-FESQ-C, HLR-FESQ-D, HLR-FESQ-E, and HLR-FESQ-F.
HLR-FPRM-B	The Internal Fire Plant Response Model shall include fire-induced initiating events, both fire-induced and random failures of equipment, fire-specific as well as non-fire-related human failures associated with safe shutdown, events in the event sequences (e.g., containment failure modes), and the supporting probability data (including uncertainty) based on the SRs stated under this High Level Requirement (HLR) that parallel, as appropriate, the internal-events PRA.
HLR-FPRM-C	The documentation of the Internal Fire Plant Response Model shall provide traceability of the work.

Table 4.3.9.5-2 Supporting Requirements for HLR-FPRM-A

The Internal Fire PRA shall include the Internal Fire Plant Response Model capable of the SRs of HLR-FESQ-A, HLR-FESQ-B, HLR-FESQ-C, HLR-FESQ-D, HLR-FESQ-E, and HLR-FESQ-F. (HLR-FPRM-A)

Index No. FPRM-A	Capability Category I	Capability Category II
FPRM-A1	CONSTRUCT the Internal Fire Plant Response Model so that it is capable of determining conditional probability of each fire-induced event sequence and event sequence family modeled in the PRA for the fire scenarios, and their associated damage target sets, defined per the Technical Subelement Internal Fire Scenario Selection and Analysis.	
FPRM-A2	CONSTRUCT the Internal Fire Plant Response Model so that it is capable of determining the frequencies of fire-induced event sequences and event sequence families modeled in the PRA once the fire frequencies (see the SRs of HLR-FIGN-A and HLR-FIGN-B) are also applied to the quantification.	
FPRM-A3	CONSTRUCT the Internal Fire Plant Response Model so that it is capable of determining the risk-significant contributors to the fire-induced risk consistent with Internal Fire Event Sequence Quantification.	

Table 4.3.9.5-3 Supporting Requirements for HLR-FPRM-B

The Internal Fire Plant Response Model shall include fire-induced initiating events, both fire-induced and random failures of equipment, fire-specific as well as non-fire-related human failures associated with safe shutdown, events in the event sequences (e.g., containment failure modes), and the supporting probability data (including uncertainty) based on the SRs stated under this HLR that parallel, as appropriate, the internal-events PRA. (HLR-FPRM-B)

Index No. FPRM-B	Capability Category I	Capability Category II
FPRM-B1	USE the event sequences and the systems logic model from the internal event PRA models as the basis of the Internal Fire Plant Response Model. INCLUDE additional event sequences, as applicable, associated with fire-induced initiating events that are not included in the internal events PRA model such as events that impact two or more reactors or sources of radioactive material. See IE-A16 See Note F-N-31	
FPRM-B2	ENSURE that the peer review findings for the internal events and other hazard PRAs that are relevant to the internal fire PRA are resolved and incorporated into the development of the Internal Fire Plant Response Model and that the disposition does not adversely affect the development of the Internal Fire Plant Response Model.	
FPRM-B3	For operating plants, CONSTRUCT the Internal Fire Plant Response Model in a manner that includes cable damage effects on the equipment of interest per Internal Fire Equipment Selection and Internal Fire Cable Selection and Location. See Note F-N-6	
FPRM-B4	For PRAs performed during the pre-operational stage, CONSTRUCT the Internal Fire Plant Response Model in a manner that includes cable damage effects on the equipment of interest per Internal Fire Equipment Selection and Internal Fire Cable Selection and Location. If exact cable information is not available, assumed cable information can be used under the requirements of Requirement FCS-A4. See Note F-N-8	
FPRM-B5	For any new initiating events identified per Requirement FES-A3, SATISFY the Capability Category I (CC-I) SRs of HLR-IE-B for the Initiating Event Analysis except where the requirements are not applicable (e.g., excluding initiating events that cannot be induced by a fire).	For any new initiating events identified per Requirement FES-A3, SATISFY the Capability Category II (CC-II) SRs of HLR-IE-B for the Initiating Event Analysis except where the requirements are not applicable (e.g., excluding initiating events that cannot be induced by a fire).
FPRM-B6	For those fire-induced initiating events included in the internal events PRA model, review the corresponding event sequence models and IDENTIFY *(a)* any existing event sequences that will require modification based on unique aspects of the plant fire response procedures; *(b)* any new event sequences that might result from a fire event that were not included in the internal events PRA; and SATISFY the CC-I SRs of HLR-ES-A and HLR-ES-B for the Event Sequence Analysis except where the requirements are not applicable.	For those fire-induced initiating events included in the internal events PRA model, review the corresponding event sequence models and IDENTIFY *(a)* any existing event sequences that will require modification based on unique aspects of the plant fire response procedures; *(b)* any new event sequences that might result from a fire event that were not included in the internal events PRA; and SATISFY the CC-II SRs of HLR-ES-A and HLR-ES-B for the Event Sequence Analysis except where the requirements are not applicable.

Table 4.3.9.5-3 Supporting Requirements for HLR-FPRM-B (Cont'd)

The Internal Fire Plant Response Model shall include fire-induced initiating events, both fire-induced and random failures of equipment, fire-specific as well as non-fire-related human failures associated with safe shutdown, events in the event sequences (e.g., containment failure modes), and the supporting probability data (including uncertainty) based on the SRs stated under this HLR that parallel, as appropriate, the internal-events PRA. (HLR-FPRM-B)

Index No. FPRM-B	Capability Category I	Capability Category II
FPRM-B7	MODEL event sequences for any new initiating events identified per Requirement FES-A3 and any event sequences identified per Requirement FPRM-B6 that represent possible plant responses to the fire-induced initiating events and SATISFY the CC-I SRs of HLR-ES-A and HLR-ES-B for the Event Sequence Analysis except where the requirements are not applicable with the following clarifications: *(a)* all the SRs of HLR-ES-A and HLR-ES-B are to be addressed in the context of fire scenarios; *(b)* when applying Requirement ES-A5 to Internal Fire PRA, INCLUDE fire response procedures as well as emergency operating procedures and abnormal procedures.	MODEL event sequences for any new initiating events identified per Requirement FES-A3 and any event sequences identified per Requirement FPRM-B6 that represent possible plant responses to the fire-induced initiating events and SATISFY the CC-II SRs of HLR-ES-A and HLR-ES-B for the Event Sequence Analysis except where the requirements are not applicable with the following clarifications: *(a)* all the SRs of HLR-ES-A and HLR-ES-B are to be addressed in the context of fire scenarios; *(b)* when applying Requirement ES-A5 to Internal Fire PRA, INCLUDE fire response procedures as well as emergency operating procedures and abnormal procedures.
FPRM-B8	IDENTIFY any cases where new or modified success criteria will be needed to support the Internal Fire PRA and SATISFY the CC-I SRs of HLR-SC-A for success criteria except where the requirements are not applicable.	IDENTIFY any cases where new or modified success criteria will be needed to support the Internal Fire PRA and SATISFY the CC-II SRs of HLR-SC-A for success criteria except where the requirements are not applicable.
FPRM-B9	DEFINE any new or modified success criteria identified per Requirement FPRM-B8 and SATISFY the CC-I SRs of HLR-SC-B for success criteria except where the requirements are not applicable.	DEFINE any new or modified success criteria identified per Requirement FPRM-B8 and SATISFY the CC-II SRs of HLR-SC-B for success criteria except where the requirements are not applicable.

Table 4.3.9.5-3 Supporting Requirements for HLR-FPRM-B (Cont'd)

The Internal Fire Plant Response Model shall include fire-induced initiating events, both fire-induced and random failures of equipment, fire-specific as well as non-fire-related human failures associated with safe shutdown, events in the event sequences (e.g., containment failure modes), and the supporting probability data (including uncertainty) based on the SRs stated under this HLR that parallel, as appropriate, the internal-events PRA. (HLR-FPRM-B)

Index No. FPRM-B	Capability Category I	Capability Category II
FPRM-B10	For any cases where new system models or split fractions are needed, or existing models or split fractions need to be modified, INCLUDE in the Internal Fire Plant Response Model the effects of the following: *(a)* fire-induced equipment failures; *(b)* fire-specific operator actions as identified per Internal Fire Human Reliability Analysis; *(c)* fire-induced spurious operations as identified per Internal Fire Equipment Selection and Internal Fire Cable Selection and Location. SATISFY the CC-I SRs of HLR-SY-A and HLR-SY-B for Systems Analysis except where the requirements are not applicable with the following clarification: All the SRs of HLR-SY-A and HLR-SY-B are to be addressed in the context of fire scenarios, including effects on system operability/functionality and including fire damage to equipment and associated cabling.	For any cases where new system models or split fractions are needed, or existing models or split fractions need to be modified, INCLUDE in the Internal Fire Plant Response Model the effects of the following: *(a)* fire-induced equipment failures; *(b)* fire-specific operator actions as identified per Internal Fire Human Reliability Analysis; *(c)* fire-induced spurious operations as identified per Internal Fire Equipment Selection and Internal Fire Cable Selection and Location. SATISFY the CC-II SRs of HLR-SY-A and HLR-SY-B for Systems Analysis except where the requirements are not applicable with the following clarification: All the SRs of HLR-SY-A and HLR-SY-B are to be addressed in the context of fire scenarios, including effects on system operability/functionality and including fire damage to equipment and associated cabling.
FPRM-B11	MODIFY the Internal Fire Plant Response Model so that systems and equipment included in the internal events PRA that are potentially vulnerable to fire-induced failure are failed in the most conservative mode consistent with the applicable event sequences, including fire-induced spurious operation, if *(a)* the cables have not been routed as per Requirement FCS-A2; and *(b)* the cables have not been routed by assumption (i.e., see Requirement FCS-A4).	
FPRM-B12	IDENTIFY any fire PRA Internal Fire Plant Response Model probability input values that either require reanalysis given the fire context or that were not included in the internal events PRA. EXCLUDE from this requirement any parameters specific to Internal Fire Scenario Selection and Analysis, Internal Fire Ignition Frequency, Internal Fire Circuit Failure Analysis and Internal Fire Human Reliability Analysis.	
FPRM-B13	For any item identified per Requirement FPRM-B12, PERFORM the Data Analysis and SATISFY the CC-I SRs of HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D for Data Analysis except where the requirements are not applicable with the following clarification: all the SRs for HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D are to be addressed in the context of both random events as well as fire events causing damage to equipment and associated cabling.	For any item identified per Requirement FPRM-B12, PERFORM the Data Analysis and SATISFY the CC-II SRs of HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D for Data Analysis except where the requirements are not applicable with the following clarification: all the SRs of HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D are to be addressed in the context of both random events as well as fire events causing damage to equipment and associated cabling.
FPRM-B14	IDENTIFY any new event sequences that would apply to the internal fire PRA that were not addressed in the PRA models for internal initiating events.	

Table 4.3.9.5-3 Supporting Requirements for HLR-FPRM-B (Cont'd)

The Internal Fire Plant Response Model shall include fire-induced initiating events, both fire-induced and random failures of equipment, fire-specific as well as non-fire-related human failures associated with safe shutdown, events in the event sequences (e.g., containment failure modes), and the supporting probability data (including uncertainty) based on the SRs stated under this HLR that parallel, as appropriate, the internal-events PRA. (HLR-FPRM-B)

Index No. FPRM-B	Capability Category I / Capability Category II
FPRM-B15	MODEL any new event sequences identified per Requirement FPRM-B14 to determine the fire-induced event sequences and event sequence families in accordance with the SRs of HLR-FES-C with the following clarifications: *(a)* all the SRs of HLR-FES-C are to be addressed in the context of fire scenarios; *(b)* requirements for Human Reliability Analysis are to be met consistent with requirements for Internal Fire Human Reliability Analysis; *(c)* requirements for Systems Analysis are to be met consistent with Requirement FPRM-B11; *(d)* requirements for Event Sequence Analysis are to be met consistent with Requirement FPRM-B7.
FPRM-B16	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the Internal Fire Plant Response Model analysis in a manner that supports Requirement FESQ-E1.
FPRM-B17	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Internal Fire Plant Response Model analysis. See Note F-N-7, F-N-8

Table 4.3.9.5-4 Supporting Requirements for HLR-FPRM-C

The documentation of the Internal Fire Plant Response Model analysis shall provide traceability of the work. (HLR-FPRM-C)

Index No. FPRM-C	Capability Category I / Capability Category II
FPRM-C1	DOCUMENT the process used in the Internal Fire Plant Response Model analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the disposition of internal events PRA peer review exceptions and deficiencies for the Internal Fire PRA; *(b)* the basis for the fire-induced initiating events included in the plant response model; *(c)* the basis for modeling of event sequence progressions that are added per Requirements FPRM-B7 and FPRM-B15; *(d)* any modification performed in the internal events model logic, including added or modified initiating events, data, success criteria, and event sequences, to represent fire-induced scenarios in the plant response model.
FPRM-C2	DOCUMENT the Internal Fire Plant Response Model and SATISFY the documentation SRs of HLR-IE-D for the Initiating Event Analysis, HLR-ES-C for the Event Sequence Analysis, HLR-SC-C for success criteria, HLR-SY-C for Systems Analysis, and HLR-DA-E for Data Analysis as well as requirements for Internal Fire Circuit Failure Analysis, with the following clarification except where the requirements are not applicable: SRs of HLR-IE-D is to be met in a manner consistent with the SRs of HLR-FIGN-B of this Section.
FPRM-C3	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FPRM-B16) associated with the Internal Fire Plant Response Model analysis.
FPRM-C4	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Internal Fire Plant Response Model analysis. See FPRM-B17 See Note F-N-8

4.3.9.6 Objectives and Technical Requirements for Internal Fire Scenario Selection and Analysis (FSS)

The objectives of the Internal Fire Scenario Selection and Analysis ensure that

(a) a set of fire scenarios are selected for each physical analysis unit that has not been screened out upon which fire risk estimates will be based;

(b) scenarios are included where primary command and control is transferred outside the main control room, if applicable;

(c) selected fire scenarios are characterized;

(d) fire analysis tools are selected and applied properly;

(e) the conditional probabilities of target damage are quantified;

(f) ignition sources that can induce failure of steel are selected and analyzed;

(g) multicompartment failures are identified;

(h) the Internal Fire Scenario Selection and Analysis is documented to provide traceability of the analysis.

Table 4.3.9.6-1 High Level Requirements for Internal Fire Scenario Selection and Analysis

Designator	Requirement
HLR-FSS-A	The Internal Fire PRA shall select sufficient combinations of an ignition source (or group of ignition sources) and damage target sets to represent the fire scenarios for each PAU that has not been screened out and upon which an estimation of the risk contribution (i.e., contribution to event sequence and event sequence family frequency) will be based.
HLR-FSS-B	The Internal Fire PRA shall include an analysis of potential fire scenarios, leading to the transfer of primary command and control outside the main control room.
HLR-FSS-C	The Internal Fire PRA shall characterize the factors that will influence the timing and extent of fire damage for each combination of an ignition source and damage target sets selected per the SRs of HLR-FSS-A.
HLR-FSS-D	The Internal Fire PRA shall select and apply appropriate fire analysis tools.
HLR-FSS-E	The Internal Fire PRA shall quantify the conditional probabilities of target damage given fire ignition.
HLR-FSS-F	The Internal Fire PRA shall search for and analyze risk-relevant ignition sources with the potential for causing fire-induced failure of exposed structural steel.
HLR-FSS-G	The Internal Fire PRA shall identify multi-compartment fire scenarios for which the risk contribution will be estimated.
HLR-FSS-H	The documentation of the Internal Fire Scenario Selection and Analysis shall provide traceability of the work.

Table 4.3.9.6-2 Supporting Requirements for HLR-FSS-A

The Internal Fire PRA shall select sufficient combinations of an ignition source (or group of ignition sources) and damage target sets to represent the fire scenarios for each PAU that has not been screened out and upon which an estimation of the risk contribution (i.e., contribution to event sequence and event sequence family frequency) will be based. (HLR-FSS-A)

Index No. FSS-A	Capability Category I	Capability Category II
FSS-A1	IDENTIFY the ignition sources, both fixed and transient, in each PAU that has not been screened out within the global analysis boundary that are capable of creating fire-induced environmental conditions, including through fire spread, that can cause the failure of at least one internal fire PRA equipment item or cable (i.e., a risk-relevant damage target).	
FSS-A2	IDENTIFY risk-relevant damage targets in each PAU that has not been screened out within the global analysis boundary.	
FSS-A3	If the exact routing of a cable (or group of cables) has not been established (see Requirements FCS-A3 and FCS-A4), ASSUME that those cables fail for any fire scenario that has a damaging effect on any raceway or conduit where the subject cable cannot be excluded.	
FSS-A4	For each PAU that has not been screened out within the global analysis boundary, SELECT sufficient combinations of a fire ignition source (or group of ignition sources) and target sets as characteristics of the selected fire scenarios so that the fire risk contribution can be characterized commensurate with whether it is a risk-significant contributor. See Note F-N-32, F-N-33	

Table 4.3.9.6-3 Supporting Requirements for HLR-FSS-B

The Internal Fire PRA shall include an analysis of potential fire scenarios, leading to the transfer of primary command and control outside the main control room. (HLR-FSS-B)

Index No. FSS-B	Capability Category I	Capability Category II
FSS-B1	SPECIFY and JUSTIFY the conditions that are assumed to require a transfer of primary command and control outside the main control room. INCLUDE both control room habitability issues and loss of control room control functions.	
FSS-B2	SELECT a sufficient number of fire scenarios, either in the control room or elsewhere, leading to a transfer of primary command and control outside the main control room so that the fire risk contribution of control room abandonment can be bounded.	SELECT a sufficient number of fire scenarios, either in the control room or elsewhere, leading to a transfer of primary command and control outside the main control room so that the fire risk contribution of control room abandonment *(a)* can be characterized; *(b)* is correlated to specific ignition sources and target sets for risk-significant contributors.

Table 4.3.9.6-4 Supporting Requirements for HLR-FSS-C

The Internal Fire PRA shall characterize the factors that will influence the timing and extent of fire damage for each combination of an ignition source and damage target sets selected per the SRs of HLR-FSS-A. (HLR-FSS-C)

Index No. FSS-C	Capability Category I	Capability Category II
FSS-C1	For fire scenarios selected in accordance with the SRs of HLR-FSS-A and HLR-FSS-B, ASSIGN intensity and duration characteristics to the ignition sources that are conservative or bounding.	For ignition sources that are risk-significant contributors and where supported by the current state-of-practice, PROVIDE a probabilistic representation of *(a)* the effects of ignition source type and location; *(b)* the range of fire heat release rate profiles; *(c)* the contribution of low-likelihood but potentially more challenging fires. For fire scenarios that are risk-significant contributors where a probabilistic representation of the ignition source is not available, SPECIFY the basis for the characterization of the fire ignition source used in the analysis.
FSS-C2	CHARACTERIZE ignition source intensity such that the fire is initiated at full-peak intensity (i.e., heat release rate).	For those scenarios that are risk-significant contributors, CHARACTERIZE ignition source intensity using a time-dependent fire growth profile (i.e., a time-dependent heat release rate) representative of the ignition source.
FSS-C3	CHARACTERIZE the total heat release rate profile of the fire source and ignition of secondary combustibles including, as appropriate to the scenario and its risk significance, fire growth, steady burning and decay stages, as appropriate to the scenario and consistent with its risk significance.	

Table 4.3.9.6-4 Supporting Requirements for HLR-FSS-C (Cont'd)

The Internal Fire PRA shall characterize the factors that will influence the timing and extent of fire damage for each combination of an ignition source and damage target sets selected per the SRs of HLR-FSS-A. (HLR-FSS-C)

Index No. FSS-C	Capability Category I	Capability Category II
FSS-C4	If a severity factor is applied in the analysis, ENSURE that *(a)* the severity factor remains independent of other quantification factors; *(b)* the event set is the same as the set used to estimate fire frequency for any severity factor relying on event data; *(c)* the severity factor applied is based on the conservative or bounding conditions and assumptions that could influence whether or not a fire will damage targets for the specific set of fire scenarios to which the severity factor is applied; *(d)* a basis supporting the severity factor's determination is stated. See Note F-N-34	APPLY severity factors for fire scenarios that are risk-significant contributors. ENSURE that *(a)* the severity factor remains independent of other quantification factors; *(b)* if the severity factor relies on insights from event data, the event set is the same as the set used to estimate fire frequency; *(c)* the severity factor takes into account the conditions and assumptions that could influence whether or not a fire will damage targets of the specific fire scenario under analysis; *(d)* a basis supporting the severity factor's determination is stated. See Note F-N-34
FSS-C5	JUSTIFY that the damage criteria used in the internal fire PRA are representative of the damage targets associated with each fire scenario.	
FSS-C6	ASSUME target damage occurs when the exposure environment exceeds the damage threshold.	For fire scenarios that are risk-significant contributors where target thermal response analysis would make a material difference to risk estimates, CALCULATE target damage times based on the thermal response of the damage target.
FSS-C7	If raceway fire wraps, other passive fire barrier elements, or active fire barrier elements within a single PAU are credited in the analysis of fire scenarios: *(a)* SPECIFY a basis for their fire-resistance rating; *(b)* CONFIRM that the fire wrap or other passive fire-protection features will not be subjected to either mechanical damage or damage from direct flame impingement from a high-hazard ignition source unless the element has been subject to qualification or other proof of performance by analysis or testing under these conditions; *(c)* INCLUDE analysis of fire scenarios involving the failure of the credited fire barrier element. See Note F-N-35, F-N-36	

Table 4.3.9.6-5 Supporting Requirements for HLR-FSS-D

The Internal Fire PRA shall select and apply appropriate fire analysis tools. (HLR-FSS-D)

<table>
<tr><th>Index No. FSS-D</th><th>Capability Category I</th><th>Capability Category II</th></tr>
<tr><td>FSS-D1</td><td colspan="2">USE analytical, empirical, and/or statistical fire modeling tools that have sufficient capability to model the conditions of interest and only within known limits of applicability.</td></tr>
<tr><td>FSS-D2</td><td>USE conservative assumptions regarding the likelihood and/or extent of fire damage in the analysis of fire scenarios such that the risk significance of each PAU, which is not screened out, is bounded.</td><td>For each fire scenario that is not screened out, APPLY fire analysis tools sufficient to characterize the risk significance of the fire scenario.</td></tr>
<tr><td>FSS-D3</td><td colspan="2">SPECIFY a basis for fire modeling tool input values used in the analysis given the context of the fire scenarios being analyzed.
See Note F-N-37</td></tr>
<tr><td>FSS-D4</td><td colspan="2">For any fire modeling parameters not covered by SRs of HLR-FSS-C, USE plant-specific parameter estimates for fire modeling if available; otherwise, use generic information.</td></tr>
<tr><td>FSS-D5</td><td colspan="2">If neither plant-specific data nor generic parameter estimates are available for fire modeling, USE parameter values for the most similar situation, adjusting if necessary, to account for differences;
or
SATISFY the requirements of Section 4.2, Use of Expert Judgment, and SPECIFY the basis for the choice of parameter values.</td></tr>
<tr><td>FSS-D6</td><td colspan="2">If statistical models are applied in the fire scenario analyses, SPECIFY a basis for the applied models.
See Note F-N-38</td></tr>
<tr><td>FSS-D7</td><td colspan="2">SPECIFY a basis for any applied empirical models in the context of the fire scenarios being analyzed by
(a) citing a referenced document; or
(b) developing the basis if
(1) one is not available in referenced documentation (e.g., technical reports describing the empirical models); or
(2) the empirical models are used outside the recommended scenario conditions.
See Note F-N-39</td></tr>
<tr><td>FSS-D8</td><td colspan="2">EVALUATE the potential for smoke damage to internal fire PRA equipment on a qualitative basis, and INCLUDE the results of this assessment in the definition of fire scenario target sets.
See Note F-N-40</td></tr>
<tr><td>FSS-D9</td><td colspan="2">EVALUATE the information on the combinations of fire sources and target sets that were selected per Requirement FSS-A4 for the as-designed, or as-built, and as-operated or as-intended-to-operate plant conditions via investigation(s), depending on the plant design-life cycle stage of the PRA to CONFIRM that these combinations represent the as-built plant conditions for those PAUs that represent risk-significant contributors.
See Note F-N-5, F-N-41</td></tr>
<tr><td>FSS-D10</td><td colspan="2">For operational plants, for PAUs that are risk-significant contributors, CONFIRM by investigations that the selected fire scenarios represent the following conditions:
(a) characteristics of the ignition source that influence fire heat release rate;
(b) the location of damage targets relative to ignition sources;
(c) proximity, type, and configuration of secondary combustibles;
(d) location, type, and physical condition of raceway fire barrier systems;
(e) placement of fixed fire detection and suppression equipment;
(f) physical and ventilation characteristics of the PAU.
See Note F-N-5, F-N-6, F-N-41</td></tr>
<tr><td>FSS-D11</td><td colspan="2">For PRAs performed during the pre-operational stage, for PAUs that are risk-significant contributors, PERFORM interviews with system designers or knowledgeable plant-design personnel to confirm that the selected fire scenarios represent the following conditions:
(a) characteristics of the ignition source that influence fire heat release rate;
(b) the location of damage targets relative to ignition sources;
(c) proximity, type, and configuration of secondary combustibles;
(d) location, type, and physical condition of raceway fire barrier systems;
(e) placement of fixed fire detection and suppression equipment;
(f) physical and ventilation characteristics of the PAU.
See Note F-N-8, F-N-41</td></tr>
</table>

Table 4.3.9.6-6 Supporting Requirements for HLR-FSS-E

The Internal Fire PRA shall quantify the conditional probabilities of target damage given fire ignition. (HLR-FSS-E)

Index No. FSS-E	Capability Category I	Capability Category II
FSS-E1	In crediting fire detection and suppression systems, USE generic estimates of total system unavailability provided that *(a)* the credited system is installed and maintained in accordance with applicable codes and standards; *(b)* the credited system is in a fully operable state during plant operation; *(c)* if multiple suppression paths are credited, dependencies among the credited paths are modeled, including dependencies associated with recovery of a failed fire suppression system, if such recovery is credited. See Note F-N-42, F-N-43, F-N-44	In crediting fire detection and suppression systems, USE generic estimates of total system unavailability provided that *(a)* the credited system is installed and maintained in accordance with applicable codes and standards; *(b)* the credited system is in a fully operable state during plant operation; *(c)* if multiple suppression paths are credited, dependencies among the credited paths are modeled, including dependencies associated with recovery of a failed fire suppression system, if such recovery is credited; *(d)* plant operating experience has been reviewed and the system has not experienced outlier behavior relative to total system unavailability. If outlier behavior relative to system unavailability is detected, CALCULATE the system unavailability and SATISFY the CC-II SRs of HLR-DA-D for Data Analysis, except where the requirements are not applicable. See Note F-N-42, F-N-43, F-N-44, F-N-45
FSS-E2	For PRAs performed during the pre-operational stage, PERFORM interviews with system designers or knowledgeable plant-design personnel to confirm that generic estimates of total system unavailability represent the as-to-be built plant. See Note F-N-8	
FSS-E3	INCLUDE an assessment of fire detection and suppression systems effectiveness in the context of each fire scenario analyzed that includes the following: *(a)* the time available to suppress the fire prior to target damage; *(b)* specific features of PAU and fire scenario under analysis (e.g., pocketing effects, blockages that might impact plume behaviors or the "visibility" of the fire to detection and suppression systems, and suppression system coverage); and *(c)* suitability of the installed system given the nature of the fire source being analyzed.	
FSS-E4	For each combination of a fire ignition source and a target set (e.g., see Requirement FSS-A4) whose analysis has taken credit for fire suppression prior to fire damage, CALCULATE a point estimate of the nonsuppression probability. For fire scenarios that are risk-significant contributors, CHARACTERIZE the uncertainty in the estimated nonsuppression probability. See Note F-N-46	For each combination of a fire ignition source and a target set (e.g., see Requirement FSS-A4) whose analysis has taken credit for fire suppression prior to fire damage, the following actions apply: *(a)* For fire scenarios that are risk-significant contributors, CALCULATE a mean value of the nonsuppression probability and PROVIDE a probabilistic representation of the uncertainty in the estimated nonsuppression probability. *(b)* For fire scenarios that are non-risk-significant contributors, CALCULATE a point estimate value of the nonsuppression probability. See Note F-N-46
FSS-E5	CONFIRM that the data used to develop the manual nonsuppression probabilities and the corresponding fire ignition frequency values (see the SRs of HLR-FIGN-A) have been used consistently so as to avoid double counting. See Note F-N-47, F-N-48	

Table 4.3.9.6-7 Supporting Requirements for HLR-FSS-F

The Internal Fire PRA shall search for and analyze risk-relevant ignition sources with the potential for causing fire-induced failure of exposed structural steel. (HLR-FSS-F)

Index No. FSS-F	Capability Category I	Capability Category II
FSS-F1	IDENTIFY any locations within the internal fire PRA global analysis boundary that meet both of the following conditions: *(a)* exposed structural steel is present; *(b)* a high-hazard fire source is present in that location. If such locations are identified, SELECT those fire scenarios that could potentially damage, including collapse, the exposed structural steel for each identified location. See Note F-N-49	
FSS-F2	If scenarios are selected per Requirement FSS-F1, PERFORM a qualitative assessment of the risk of the selected fire scenarios, including collapse of the exposed structural steel.	If, per Requirement FSS-F1, one or more scenarios are selected, SPECIFY the technical basis for the criteria associated with structural collapse due to fire exposure; and PERFORM a quantitative assessment of the risk of the selected fire scenarios in a manner consistent with the SRs of HLR-FESQ-A, HLR-FESQ-B, HLR-FESQ-C, HLR-FESQ-D, HLR-FESQ-E, and HLR-FESQ-F, including collapse of the exposed structural steel.

Table 4.3.9.6-8 Supporting Requirements for HLR-FSS-G

The Internal Fire PRA shall identify multi-compartment fire scenarios for which the risk contribution will be estimated. (HLR-FSS-G)

Index No. FSS-G	Capability Category I	Capability Category II
FSS-G1	For fire modeling of single PAU to the modeling of multi-compartment fire scenarios, SATISFY all the SRs listed in Requirements FSS-C1, FSS-C2, FSS-C3, FSS-C4, FSS-C5, FSS-C6, and FSS-C7 except where the requirements are not applicable. See Note F-N-50	
FSS-G2	For multi-compartment fire scenarios, APPLY the screening criteria per the requirements of SCR-2 and SCR-3 in Table 1.10-1 to all the PAUs within the global analysis boundaries.	
FSS-G3	For each PAU combination that is not screened out, SELECT a sufficient number of multi-compartment fire scenario(s) so that the fire risk contribution of multi-compartment fires can be characterized.	
FSS-G4	When passive fire barriers with a fire-resistance rating are credited in the internal fire PRA, ENSURE that the credit for resistance against fire-induced failure is consistent with the fire-resistance rating as demonstrated by conformance to applicable test standards. See Note F-N-51	When passive fire barriers with a fire-resistance rating are credited in the internal fire PRA, *(a)* ENSURE that the credit for resistance against fire-induced failure is consistent with the fire-resistance rating as demonstrated by conformance to applicable test standards; and *(b)* QUANTIFY the random failure probability including reliability and availability. See Note F-N-51

Table 4.3.9.6-8 Supporting Requirements for HLR-FSS-G (Cont'd)

The Internal Fire PRA shall identify multi-compartment fire scenarios for which the risk contribution will be estimated. (HLR-FSS-G)

Index No. FSS-G	Capability Category I	Capability Category II
FSS-G5	If passive fire barriers that lack a fire-resistance rating are credited in the internal fire PRA, SPECIFY the basis for the credit given for resistance against fire-induced failure.	If passive fire barriers that lack a fire-resistance rating are credited in the internal fire PRA, SPECIFY the basis for the credit given including resistance against fire-induced failure; and QUANTIFY the random failure probability including reliability and availability.
FSS-G6	For any scenario selected per Requirement FSS-G3, if the adjoining PAUs are separated by active fire barrier elements, ASSESS qualitatively the effectiveness, reliability, and availability of the active fire barrier element. See Note F-N-52	For any scenario selected per Requirement FSS-G3, if the adjoining PAUs are separated by active fire barrier elements, *(a)* CALCULATE the reliability and availability of the active fire barrier element; *(b)* CONFIRM that the active fire barrier element will be effective given the nature of the fire threat being postulated. See Note F-N-52
FSS-G7	ASSESS qualitatively the potential risk significance of any selected multi-compartment fire scenarios.	CALCULATE the risk contribution of any selected multi-compartment fire scenarios in a manner consistent with the SRs of HLR-FESQ-A, HLR-FESQ-B, HLR-FESQ-C, HLR-FESQ-D, HLR-FESQ-E, and HLR-FESQ-F.
FSS-G8	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the fire scenario selection and analysis in a manner that supports Requirement FESQ-E1.	
FSS-G9	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the fire scenario selection and analysis. See Note F-N-7, F-N-8	

Table 4.3.9.6-9 Supporting Requirements for HLR-FSS-H

The documentation of the Fire Scenario Selection and analysis shall provide traceability of the work. (HLR-FSS-H)

Index No. FSS-H	Capability Category I	Capability Category II
FSS-H1	DOCUMENT the process used in the fire scenario selection, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the basis for target damage mechanisms and thresholds used in the analysis, including references for any plant-specific or target-specific performance criteria applied in the analysis; *(b)* the basis for the selection of the applied fire modeling tools; *(c)* a basis for any statistical and empirical models applied in the analysis, including applicability; *(d)* a basis for any plant-specific updates applied to generic statistical models; *(e)* the assumptions made related to credited firefighting activities, including fire detection, fire suppression systems, and any credit given to manual suppression efforts; *(f)* the methodology used to select and quantify scenarios with the potential for causing fire-induced failure of exposed structural steel; *(g)* the methodology used to select multi-compartment fire scenarios that are potentially risk-significant contributors; *(h)* investigation process and results.	
FSS-H2	For each fire scenario, DOCUMENT the fire growth and damage analysis and related assumptions, including the following: *(a)* the nature and characteristics of the ignition source; *(b)* the nature and characteristics of the damage target set; *(c)* any applied severity factors; *(d)* the calculated nonsuppression probability; *(e)* the fire modeling tool input values used in the analysis of each fire scenario; *(f)* the fire modeling output results for each analyzed fire scenario, including the results of parameter uncertainty evaluations (as performed).	
FSS-H3	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FSS-G8) associated with the fire scenario selection.	
FSS-H4	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Fire Scenario Selection. See FSS-G9 See Note F-N-8	

4.3.9.7 Objectives and Technical Requirements for Internal Fire Ignition Frequency (FIGN)

The objectives of the Internal Fire Ignition Frequency technical element ensure that

(a) fire ignition frequencies are established for fires of various types not previously screened; and
(b) the Internal Fire Ignition Frequency selection is documented to provide traceability of the analysis.

Table 4.3.9.7-1 High-Level Requirements for Internal Fire Ignition Frequency

Designator	Requirement
HLR-FIGN-A	The Internal Fire PRA shall estimate fire ignition frequencies for every PAU that has not been qualitatively screened out.
HLR-FIGN-B	The documentation of the Ignition Frequency Analysis shall provide traceability of the work.

Table 4.3.9.7-2 Supporting Requirements for HLR-FIGN-A

The Internal Fire PRA shall estimate fire ignition frequencies for every PAU that has not been qualitatively screened out. (HLR-FIGN-A)

Index No. FIGN-A	Capability Category I	Capability Category II
FIGN-A1	If similar plants exist, except as allowed by Requirements FIGN-A2 and FIGN-A3, USE current nuclear power industry event history that includes power plants of similar type, characteristics, and vintage to establish ignition frequencies on a per plant-year basis. SPECIFY the basis for the exclusion of data judged to be non-applicable (e.g., due to changes in industry practices). See Note F-N-53	
FIGN-A2	Except as allowed by Requirement FIGN-A3, USE applicable data from nonnuclear power industry sources only when there is no similar experience in the nuclear power industry. JUSTIFY all nonnuclear power industry sources used for establishing fire ignition frequencies by demonstrating the applicability of information stated in those sources to the specific ignition source being studied. In justifying the use of nonnuclear power industry data, CONFIRM that *(a)* applicable nuclear industry data does not exist; a description of the data being applied including its source is documented; discussion of the data analysis approach and methods used to estimate per plant-year fire frequencies is documented; and the applicability of the data to nuclear power plant conditions and the fire scenario(s) being analyzed; *(b)* the underlying data set is applicable to the specific ignition source being studied; *(c)* the underlying data set is applicable to nuclear power plant conditions and the fire scenario(s) being analyzed; *(d)* the scope and completeness of the underlying data set is adequate to support robust statistical treatment; *(e)* the total population base and equivalent years of operating experience represented by the underlying data set can be quantified; *(f)* the fire frequencies calculated are consistent with, and have properly analyzed dependencies with, or are independent from other aspects of the internal fire PRA, including, in particular, any applied fire severity (e.g., fire severity factor) treatments and/or any mitigation credit applied for fire detection and suppression prior to target damage, including the analysis of both timing and effectiveness; *(g)* the underlying data set and all analyses performed are available for review.	
FIGN-A3	In cases where nuclear power industry and nonnuclear industry data are not available, SATISFY the requirements of Section 4.2, Use of Expert Judgment.	
FIGN-A4	For operating reactors, REVIEW plant-specific experience for fire event outlier experience, and PERFORM a plant-specific fire frequency update if outliers are found. See Note F-N-8, F-N-54	
FIGN-A5	For operating plants, ESTIMATE generic fire ignition frequencies or plant-specific fire frequency updates on a plant-year basis (generic fire frequencies are typically reported on this same basis). INCLUDE in the fire frequency estimation the plant availability, such that the frequencies are weighted by the fraction of time the plant is at-power. See Note F-N-6, F-N-55, F-N-56	
FIGN-A6	For PRAs performed during the pre-operational stage, ESTIMATE generic fire-ignition frequencies on a plant-year basis. See Note F-N-8, F-N-55	

Table 4.3.9.7-2 Supporting Requirements for HLR-FIGN-A (Cont'd)

The Internal Fire PRA shall estimate fire ignition frequencies for every PAU that has not been qualitatively screened out. (HLR-FIGN-A)

Index No. FIGN-A	Capability Category I	Capability Category II
FIGN-A7	When combining evidence from generic-, technology-, or plant-specific data, USE a Bayesian update process or equivalent statistical process and SATISFY the CC-I requirements in Requirement DA-D1 for Data Analysis, except where the requirements are not applicable. See Note F-N-57	When combining evidence from generic-, technology-, or plant-specific data, USE a Bayesian update process or equivalent statistical process and SATISFY the CC-II requirements in Requirement DA-D1 for Data Analysis, except where the requirements are not applicable. See Note F-N-57
FIGN-A8	USE a design- or plant-wide consistent methodology for both fixed and transient ignition sources based on parameters that are expected to influence the likelihood of ignition to apportion high-level ignition frequencies to estimate PAU or ignition source level frequencies. See Note F-N-58	
FIGN-A9	ASSIGN an ignition frequency greater than zero to every PAU that has not been qualitatively screened out. See Note F-N-59	
FIGN-A10	CALCULATE a point estimate for the ignition frequencies. CHARACTERIZE the uncertainty for those ignition frequencies associated with fire scenarios that are relevant risk-significant contributors. See Note F-N-46	CALCULATE a mean value for the ignition frequencies for the fire scenarios that are risk-significant contributors. PROVIDE a probabilistic representation of the uncertainty of the parameter estimates of ignition frequencies for the fire scenarios that are relevant risk-significant contributors. If using expert judgment in the fire ignition frequency analysis, SATISFY the requirements of Section 4.2, Use of Expert Judgment. For the fire scenarios that are non-risk-significant contributors, CALCULATE point estimates. See Note F-N-46, F-N-60
FIGN-A11	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the fire ignition frequency analysis in a manner that supports Requirement FESQ-E1.	
FIGN-A12	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the fire ignition frequency analysis. See Note F-N-7, F-N-8	

Table 4.3.9.7-3 Supporting Requirements for HLR-FIGN-B

The documentation of the Ignition Frequency Analysis shall provide traceability of the work. (HLR-FIGN-B)

Index No. FIGN-B	Capability Category I	Capability Category II
FIGN-B1	DOCUMENT the process used in the ignition frequency analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* references for fire events and fire ignition frequency sources used; *(b)* the apportioning methodology and bases of selected values; *(c)* the plant-specific frequency updating process and results, including the selected plant-specific events, the basis for the selection and or exclusion of events, the analysis supporting the plant-specific plant-years, and the Bayesian process for updating generic frequencies.	
FIGN-B2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in FIGN-A11) associated with the fire ignition frequency analysis.	
FIGN-B3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the fire ignition frequency analysis. See FIGN-A12 See Note F-N-8	

4.3.9.8 Objectives and Technical Requirements for Internal Fire Circuit Failure Analysis (FCF)

The objectives of the Internal Fire Circuit Failure Analysis technical element are to

(a) refine the understanding and analysis of fire-induced circuit failures on an individual fire scenario basis and ensure that the consequences of each fire scenario on the damaged cables and circuits have been addressed; and

(b) the Internal Fire Circuit Failure Analysis is documented to provide traceability of the analysis.

Table 4.3.9.8-1 High Level Requirements for Internal Fire Circuit Failure Analysis

Designator	Requirement
HLR-FCF-A	The Internal Fire PRA shall determine the applicable conditional probability of the cable and circuit failure mode(s) that would cause equipment functional failure and/or undesired spurious operation based on the equipment failure modes as modeled in the Internal Fire Plant Response Model.
HLR-FCF-B	The documentation of the Internal Fire Circuit Failure Analysis shall provide traceability of the work.

Table 4.3.9.8-2 Supporting Requirements for HLR-FCF-A

The Internal Fire PRA shall determine the applicable conditional probability of the cable and circuit failure mode(s) that would cause equipment functional failure and/or undesired spurious operation based on the equipment failure modes as modeled in the Internal Fire Plant Response Model. (HLR-FCF-A)

Index No. FCF-A	Capability Category I	Capability Category II
FCF-A1	ASSIGN conservative or bounding failure mode probabilities to components consistent with generic industry-wide values.	For fire scenarios that are relevant risk-significant contributors, ASSIGN the component failure mode probabilities consistent with (a) the industry-wide cable failure mode generic values, (b) the cables failed in the fire scenario, and (c) the characteristics of the damaged circuits. For fire scenarios that are relevant risk-significant contributors and that include spurious operation component failure modes that would be impacted by the consideration of hot short duration, CREDIT the mitigating effects of limited hot short duration in the analysis. For fire scenarios that are non-risk-significant contributors, ASSIGN bounding failure mode probabilities to components consistent with generic industry-wide values.
FCF-A2	CALCULATE a point estimate for the failure mode probability values assigned per Requirement FCF-A1. CHARACTERIZE the uncertainty for those probability values associated with fire scenarios that are relevant risk-significant contributors. See Note F-N-46	For the fire scenarios that are risk-significant, CALCULATE a mean value for the failure mode probability values assigned per Requirement FCF-A1 and the duration probabilities assigned per Requirement FCF-A1; and PROVIDE a probabilistic representation of the uncertainty of the parameter estimates of the probability values for the fire scenarios that are risk-significant contributors. For the fire scenarios that are non-risk-significant, CALCULATE point estimates and CHARACTERIZE the uncertainty for the failure mode probability and duration values. See Note F-N-46
FCF-A3	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the Internal Fire Circuit Failure Analysis in a manner that supports Requirement FESQ-E1.	
FCF-A4	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Internal Fire Circuit Failure Analysis. See Note F-N-7, F-N-8	

Table 4.3.9.8-3 Supporting Requirements for HLR-FCF-B

The documentation of the Internal Fire Circuit Failure Analysis shall provide traceability of the work. (HLR-FCF-B)

Index No. FCF-B	Capability Category I	Capability Category II
FCF-B1	DOCUMENT the process used in the Internal Fire Circuit Failure Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the basis for each circuit failure probability; *(b)* the basis for any hot short duration credited in the plant response model; *(c)* the uncertainty for each circuit failure probability and hot short duration probability.	
FCF-B2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in FCF-A3) associated with the Internal Fire Circuit Failure Analysis.	
FCF-B3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Internal Fire Circuit Failure Analysis. See FCF-A4 See Note F-N-8	

4.3.9.9 Objectives and Technical Requirements for Internal Fire Human Reliability Analysis (FHR)

The objectives of the Internal Fire Human Reliability Analysis technical element are to

(a) identify the post-initiator human actions to be included in the Internal Fire PRA;
(b) include events in the Internal Fire PRA associated with human actions;
(c) quantify human failure probabilities and address dependencies;
(d) include recovery actions; and
(e) the Internal Fire Human Reliability Analysis is documented to provide traceability of the analysis.

Table 4.3.9.9-1 High Level Requirements for Internal Fire Human Reliability Analysis

Designator	Requirement
HLR-FHR-A	The Internal Fire PRA shall identify new human actions relevant to the sequences in the Internal Fire Plant Response Model.
HLR-FHR-B	The Internal Fire PRA shall include events where appropriate in the Internal Fire PRA associated with any newly identified human actions.
HLR-FHR-C	The Internal Fire PRA shall quantify human error probabilities (HEPs) accounting for the plant- or design-specific and scenario-specific influences on human performance, particularly including the effects of fires, and address potential dependencies.
HLR-FHR-D	The Internal Fire PRA shall include recovery actions only if it has been demonstrated that the action is plausible and feasible for those scenarios to which it applies, particularly accounting for the effects of fires.
HLR-FHR-E	The documentation of the Internal Fire Human Reliability Analysis shall provide traceability of the work.

Table 4.3.9.9-2 Supporting Requirements for HLR-FHR-A

The Internal Fire PRA shall identify new human actions relevant to the sequences in the Internal Fire Plant Response Model. (HLR-FHR-A)

Index No. FHR-A	Capability Category I	Capability Category II
FHR-A1	For each fire scenario, IDENTIFY any new fire-specific safe shutdown actions, and SATISFY SRs of HLR-HR-E for Human Reliability Analysis except where the requirements are not applicable with the following clarifications: *(a)* where Requirement HR-E1 discusses procedures, it is to be extended to procedures for responding to fires; *(b)* where Requirement HR-E1 mentions "in the context of the scenarios," specific attention is to be given to the fact that these are fire scenarios. See Note F-N-61	For each fire scenario, IDENTIFY any new fire-specific safe shutdown actions, and SATISFY SRs of HLR-HR-E for Human Reliability Analysis except where the requirements are not applicable with the following clarifications: *(a)* where Requirement HR-E1 discusses procedures, it is to be extended to procedures for responding to fires; *(b)* where Requirement HR-E1 mentions "in the context of the scenarios," specific attention is to be given to the fact that these are fire scenarios. For fire scenarios, IDENTIFY any new, undesired operator actions that could result from spurious indications resulting from fire-induced failure of a single instrument, per Requirement FES-C2. See Note F-N-61
FHR-A2	For operating plants, REVIEW the interpretation of the procedures associated with actions identified in Requirement FHR-A1 with plant operations or training personnel to confirm that the interpretation is consistent with plant operational and training practices. See Note F-N-6	For operating plants, USE talk-throughs (i.e., review in detail) with plant operations and training personnel to confirm that the interpretation of the procedures relevant to actions identified in Requirements FHR-A1 is consistent with plant operational and training practices. See Note F-N-6
FHR-A3	For PRAs performed during the pre-operational stage, REVIEW assumptions on procedures with engineering personnel to confirm consistency with planned design and planned plant operating and training practices relevant to actions identified in Requirement FHR-A1. See Note F-N-8	For PRAs performed during the pre-operational stage, USE talk-throughs (i.e., review in detail) with plant designers to confirm that the interpretation of operating philosophies relevant to actions identified in Requirement FHR-A1 is consistent with the operating philosophies. See Note F-N-8

Table 4.3.9.9-3 Supporting Requirements for HLR-FHR-B

The Internal Fire PRA shall include events where appropriate in the Internal Fire PRA associated with any newly identified human actions. (HLR-FHR-B)

Index No. FHR-B	Capability Category I	Capability Category II
FHR-B1	INCLUDE new fire-related safe shutdown human failure events (HFEs) corresponding to the actions identified per the CC-I Requirement FHR-A1 in the Internal Fire Plant Response Model; and SATISFY the CC-I SRs of HLR-HR-F for Human Reliability Analysis except where the requirements are not applicable.	INCLUDE new fire-related safe shutdown HFEs corresponding to the actions identified per the CC-II Requirement FHR-A1 in the Internal Fire Plant Response Model; and SATISFY the CC-II SRs of HLR-HR-F for Human Reliability Analysis except where the requirements are not applicable.
FHR-B2	DEFINE the internal fire PRA HFEs, including both those retained from the internal-events analysis and those identified per Requirement FHR-B1; and SATISFY the CC-I requirements in HR-F4 for Human Reliability Analysis, except where the requirements are not applicable. See Note F-N-62	DEFINE the internal fire PRA HFEs, including both those retained from the internal-events analysis and those identified per Requirement FHR-B1; and SATISFY the CC-II requirements in HR-F4 for Human Reliability Analysis, except where the requirements are not applicable. See Note F-N-62

Table 4.3.9.9-4 Supporting Requirements for HLR-FHR-C

The Internal Fire PRA shall quantify HEPs accounting for the plant- or design-specific and scenario-specific influences on human performance, particularly including the effects of fires, and address potential dependencies. (HLR-FHR-C)

Index No. FHR-C	Capability Category I	Capability Category II
FHR-C1	CALCULATE the HEPs for all HFEs by addressing relevant fire-related effects using conservative estimates (e.g., screening values). For the calculation of HEPs, SATISFY the CC-I SRs of HLR-HR-G for Human Reliability Analysis except where the requirements are not applicable, with the following clarification: Attention is to be given to how the fire situation alters any previous assessments in non-fire analyses as to the influencing factors and the timing considerations covered in Requirements HR-G1, HR-G4, HR-G6, and HR-G8. See Note F-N-63	CALCULATE the HEPs for all HFEs by addressing relevant fire-related effects using detailed analyses for HFEs that are relevant risk-significant contributors and conservative estimates (e.g., screening values) for the remaining HEPs. For the calculation of HEPs, SATISFY the CC-II SRs of HLR-HR-G for Human Reliability Analysis except where the requirements are not applicable, with the following clarification: Attention is to be given to how the fire situation alters any previous assessments in non-fire analyses as to the influencing factors and the timing considerations covered in Requirements HR-G1, HR-G4, HR-G6, and HR-G8. See Note F-N-63

Table 4.3.9.9-5 Supporting Requirements for HLR-FHR-D

The Internal Fire PRA shall include recovery actions only if it has been demonstrated that the action is plausible and feasible for those scenarios to which it applies, particularly accounting for the effects of fires. (HLR-FHR-D)

Index No. FHR-D	Capability Category I	Capability Category II
FHR-D1	IDENTIFY fire-specific recovery actions and SATISFY Requirement HR-H1; and QUANTIFY the corresponding HEP values including relevant fire-related effects, including any effects that may preclude a recovery action or alter the manner in which it is accomplished and SATISFY Requirements HR-H2 and HR-H4 for Human Reliability Analysis except where the requirements are not applicable. See Note F-N-64	
FHR-D2	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the Internal Fire Human Reliability Analysis in a manner that supports Requirement FESQ-E1.	
FHR-D3	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Internal Fire Human Reliability Analysis. See Note F-N-7, F-N-8	

Table 4.3.9.9-6 Supporting Requirements for HLR-FHR-E

The documentation of the Internal Fire Human Reliability Analysis shall provide traceability of the work. (HLR-FHR-E)

Index No. FHR-E	Capability Category I	Capability Category II
FHR-E1	DOCUMENT the process used in the Internal Fire Human Reliability Analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the treatment of plant- or design-specific and scenario-specific influences on human reliability, particularly including the effects of fires; *(b)* new human actions and recovery actions modeled in the internal fire PRA; *(c)* the identification and quantification of the HFEs/HEPs; and Meet the SRs of HLR-HR-I, except where the requirements are not applicable.	
FHR-E2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FHR-D2) associated with the Internal Fire Human Reliability Analysis.	
FHR-E3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Internal Fire Human Reliability Analysis. See FHR-D3 See Note F-N-8	

4.3.9.10 Objectives and Technical Requirements for Internal Fire Event Sequence Quantification (FESQ)

The objectives of the Internal Fire Event Sequence Quantification technical element are to

(a) quantify the fire-induced event sequences;
(b) use appropriate models and codes appropriately;
(c) identify dependencies in quantification process;
(d) review results and risk-significant contributors;
(e) characterize uncertainties in the Internal Fire PRA; and
(f) ensure the Internal Fire Event Sequence Quantification is documented to provide traceability of the analysis.

Table 4.3.9.10-1 High Level Requirements for Internal Fire Event Sequence Quantification

Designator	Requirement
HLR-FESQ-A	The frequency of each modeled fire-induced event sequence and event sequence family shall be quantified.
HLR-FESQ-B	The fire-induced event sequence quantification shall use appropriate models and codes, a truncation level sufficiently low to show convergence, and shall include method-specific limitations and features.
HLR-FESQ-C	Model quantification shall determine that all identified dependencies are addressed appropriately.
HLR-FESQ-D	The fire-induced event sequence quantification results shall be reviewed for correctness, completeness, and consistency. The risk-significant contributors to event sequences, such as fires and their corresponding plant initiating events, fire locations, basic events (equipment unavailabilities and HFEs), plant damage states, radionuclide transport barrier challenges, and failure modes, shall be identified. The results shall be traceable to the inputs and assumptions made in the Internal Fire PRA.
HLR-FESQ-E	Uncertainties in the Internal Fire PRA results shall be characterized. The potential impact of sources of model uncertainty and related assumptions on the results shall be understood.
HLR-FESQ-F	The documentation of the Internal Fire Event Sequence Quantification shall provide traceability of the work.

Table 4.3.9.10-2 Supporting Requirements for HLR-FESQ-A

The frequency of each modeled fire-induced event sequence and event sequence family shall be quantified. (HLR-FESQ-A)

Index No. FESQ-A	Capability Category I	Capability Category II
FESQ-A1	If quantitative screening is performed, SATISFY the event sequence family screening criteria of SCR-2 in Table 1.10-1 to screen out internal fire scenarios from the Internal Fire PRA model (i.e. quantitative screening).	
FESQ-A2	For each fire scenario that will be quantified as a contributor to fire-induced risk, MODEL the equipment and cable failures as basic events or as impacts on existing basic events in the Internal Fire Plant Response Model. See Note F-N-65, F-N-66	
FESQ-A3	For each fire scenario that will be quantified as a contributor to fire-induced risk, IDENTIFY the corresponding initiating event or events in the Internal Fire Plant Response Model [e.g., general transient, loss of off-site power (LOOP)]; and JUSTIFY the selection based on the fire-induced damage associated with the fire scenario.	
FESQ-A4	For each fire scenario that will be quantified as a contributor to fire-induced risk, INCLUDE the scenario-specific quantification factors (i.e., the factors obtained per Internal Fire Circuit Failure Analysis and Internal Fire Human Reliability Analysis, and Requirements FESQ-A2 and FESQ-A3) in the Internal Fire Plant Response Model.	
FESQ-A5	CALCULATE the fire-induced risk and SATISFY the CC-I SRs of HLR-ESQ-A for quantification except where the requirements are not applicable with the following clarifications: *(a)* quantification is to include the fire ignition frequency per Internal Fire Ignition Frequency and fire-specific conditional damage probability factors per the SRs of HLR-FSS-A, HLR-FSS-B, HLR-FSS-C, HLR-FSS-D, HLR-FSS-E, HLR-FSS-F, HLR-FSS-G and HLR-FSS-H. *(b)* quantification is to include the quantification factors per Requirement FESQ-A4. *(c)* Requirement ESQ-A7 is to be met based on meeting SRs of HLR-FHR-D.	CALCULATE the fire-induced risk and SATISFY the CC-II SRs of HLR-ESQ-A for quantification except where the requirements are not applicable with the following clarifications: *(a)* quantification is to include the fire ignition frequency per Internal Fire Ignition Frequency and fire-specific conditional damage probability factors per the SRs of HLR-FSS-A, HLR-FSS-B, HLR-FSS-C, HLR-FSS-D, HLR-FSS-E, HLR-FSS-F, HLR-FSS-G, and HLR-FSS-H. *(b)* quantification is to include the quantification factors per Requirement FESQ-A4. *(c)* Requirement ESQ-A7 is to be met based on meeting SRs of HLR-FHR-D.

Table 4.3.9.10-3 Supporting Requirements for HLR-FESQ-B

The fire-induced event sequence quantification shall use appropriate models and codes, a truncation level sufficiently low to show convergence, and shall include method-specific limitations and features. (HLR-FESQ-B)

Index No. FESQ-B	Capability Category I	Capability Category II
FESQ-B1	PERFORM the quantification and SATISFY the CC-I SRs of HLR-ESQ-B for quantification except where the requirements are not applicable.	PERFORM the quantification and SATISFY the CC-II SRs of HLR-ESQ-B for quantification except where the requirements are not applicable.

Table 4.3.9.10-4 Supporting Requirements for HLR-FESQ-C

Model quantification shall determine that all identified dependencies are addressed appropriately. (HLR-FESQ-C)

Index No. FESQ-C	Capability Category I	Capability Category II
FESQ-C1	INCLUDE dependencies during the Internal Fire Plant Response Model quantification and SATISFY the CC-I SRs of HLR-ESQ-C for quantification except where the requirements are not applicable.	INCLUDE dependencies during the Internal Fire Plant Response Model quantification and SATISFY the CC-II SRs of HLR-ESQ-C for quantification except where the requirements are not applicable.

Table 4.3.9.10-5 Supporting Requirements for HLR-FESQ-D

The fire-induced event sequence quantification results shall be reviewed for correctness, completeness, and consistency. The risk-significant contributors to event sequences, such as fires and their corresponding plant initiating events, fire locations, basic events (equipment unavailabilities and HFEs), plant damage states, radionuclide transport barrier challenges, and failure modes, shall be identified. The results shall be traceable to the inputs and assumptions made in the Internal Fire PRA. (HLR-FESQ-D)

Index No. FESQ-D	Capability Category I	Capability Category II
FESQ-D1	IDENTIFY risk-significant contributors and SATISFY the CC-I SRs of HLR-ESQ-D except where the requirements are not applicable with the following clarifications: *(a)* CC-I Requirements ESQ-D6 and ESQ-D7 are to be met including identification of which fire scenarios and which PAUs (consistent with the level of resolution of the internal fire PRA such as fire area or fire compartment) are risk-significant contributors; *(b)* Requirement ESQ-D7 is to be met recognizing that "component" is generally equivalent to "equipment"; *(c)* CC-I requirement in Requirement ESQ-D4 for comparison to similar plants is not applicable. See Note F-N-67	IDENTIFY risk-significant contributors and SATISFY the CC-II SRs of HLR-ESQ-D except where the requirements are not applicable with the following clarifications: *(a)* CC-II Requirements ESQ-D6 and ESQ-D7 are to be met including identification of which fire scenarios and which PAUs (consistent with the level of resolution of the internal fire PRA such as fire area or fire compartment) are risk-significant contributors; *(b)* Requirement ESQ-D7 is to be met recognizing that "component" is generally equivalent to "equipment"; *(c)* CC-II Requirement ESQ-D4 for comparison to similar plants is not applicable. See Note F-N-67
FESQ-D2	IDENTIFY the sources of model uncertainty and assumptions associated with the Internal Fire Event Sequence Quantification analysis in a manner that supports Requirement FESQ-E1.	
FESQ-D3	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Internal Fire Event Sequence Quantification analysis. See Note F-N-7, F-N-8	

Table 4.3.9.10-6 Supporting Requirements for HLR-FESQ-E

Uncertainties in the Internal Fire PRA results shall be characterized. The potential impact of sources of model uncertainty and related assumptions on the results shall be understood. (HLR-FESQ-E)

Index No. FESQ-E	Capability Category I	Capability Category II
FESQ-E1	SATISFY Requirement ESQ-E1 for each internal fire technical subelement (as identified in Requirements FPP-B7, FES-C2, FQLS-A5, FPRM-B16, FSS-G8, FIGN-A11, FCF-A3, FHR-D2, and FESQ-D2).	
FESQ-E2	PERFORM an uncertainty analysis for the internal fire PRA and SATISFY the CC-I Requirement ESQ-E2.	PERFORM an uncertainty analysis for the internal fire PRA and SATISFY the CC-II Requirement ESQ-E2.

Table 4.3.9.10-7 Supporting Requirements for HLR-FESQ-F

The documentation of the Internal Fire Event Sequence Quantification shall provide traceability of the work. (HLR-FESQ-F)

Index No. FESQ-F	Capability Category I	Capability Category II
FESQ-F1	DOCUMENT the process used in the Internal Fire Event Sequence Quantification analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* identification of fire scenarios and PAUs that are risk-significant contributors; and *(b)* meet the documentation requirements in Requirement ESQ-F1 and Requirement ESQ-F2 except where the requirements are not applicable.	
FESQ-F2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement FESQ-D2) associated with the Internal Fire Event Sequence Quantification analysis.	
FESQ-F3	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Internal Fire Event Sequence Quantification analysis. See FESQ-D3 See Note F-N-8	
FESQ-F4	DOCUMENT limitations in the quantification process that would impact applications.	

4.3.9.11 Peer Review Requirements for Internal Fire PRA

4.3.9.11.1 Purpose

This Section provides requirements for peer review of the Internal Fire PRA.

4.3.9.11.2 Peer Review Team Composition and Personnel Qualifications

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the areas of systems engineering, PRA (including internal fire PRA and internal fire HRA), Titles 10 of the Code of Federal Regulations Part 50 Appendix R (or equivalent) Fire Safe Shutdown Analysis, circuit failure analyses, fire modeling, and fire protection programs and their elements, as applicable to the scope of the review.

4.3.9.11.3 Review of Internal Fire PRA Technical Elements to Confirm the Methodology

4.3.9.11.3.1 Internal Fire Plant Boundary Definition and Partitioning

A review shall be performed on the Internal Fire Plant Boundary Definition and Partitioning analysis. The Internal Fire Plant Boundary Definition and Partitioning analysis verification typically includes the following:

(a) the global analysis boundary is appropriate to the overall internal fire PRA scope and the intended internal fire PRA applications;

(b) the criteria used to partition the plant into physical analysis units are defined and appropriate;

(c) all fire areas within the global analysis boundary have been clearly identified;

(d) in those cases where a fire barrier that lacks a fire resistance rating or spatial separation has been credited as a partitioning feature, use a selective review to demonstrate that an appropriate multi-compartment fire scenario analysis has been conducted;

(e) a selective review by investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable] is recommended to confirm the plant partitioning analysis.

The review of the Internal Fire Plant Boundary Definition and Partitioning analysis shall be closely coordinated with the review of the corresponding multi-compartment analysis.

4.3.9.11.3.2 Internal Fire Equipment Selection

A review shall be performed on the Internal Fire Equipment Selection process. The verification typically includes the following:

(a) given the unique context of fire-induced plant damage states, the internal fire PRA has; dispositioned the initiating events included in the internal events PRA for relevance to the internal fire PRA; has reconsidered initiating events screened out from the internal events PRA for relevance to the internal fire PRA; and has identified and dispositioned potential new initiating events not considered in the internal events analysis but that might occur as the result of fire-*induced damage;*

(b) the equipment selection process has included the potentially risk-significant SSCs and their failure modes sufficient to meet the needs of the internal fire PRA application;

(c) the functions needed to support human actions in the internal fire PRA have been identified in a manner consistent with the internal fire PRA Capability Category being addressed (or otherwise that the internal fire PRA has assumed the worst failure mode for any equipment not included in the internal fire PRA).

4.3.9.11.3.3 Internal Fire Cable Selection and Location

A review shall be performed on the Internal Fire Cable Selection and Location process. The Internal Fire Cable Selection and Location process verification typically includes the following:

(a) the cable selection process is consistent with the equipment selection and associated failure modes and includes other support equipment (including locations) needed to provide the functions included in the Internal Fire Plant Response Model;

(b) the power supply and distribution systems have been analyzed in the cable selection process including fuse/breaker coordination;

(c) the cable location information (including cable endpoint location) is of sufficient depth and scope so as to support the intended internal fire PRA applications and is consistent with the physical analysis units as defined by Internal Fire Plant Boundary Definition and Partitioning.

(d) the internal fire PRA has appropriately analyzed those instances where specific cable location information is lacking.

4.3.9.11.3.4 Internal Fire Qualitative Screening

If an Internal Fire Qualitative Screening analysis has been performed, the peer review shall be performed on it. The Internal Fire Qualitative Screening analysis verification typically includes the following:

(a) appropriate qualitative screening criteria have been established;

(b) the criteria have been uniformly applied, and a justification is stated for any physical analysis units screened out of the analysis with assurance that the screening process does not cause a risk-significant contributor to be missed;

(c) a disposition has been documented for all physical analysis units within the global analysis boundary.

4.3.9.11.3.5 Internal Fire Plant Response Model

A review shall be performed on the Internal Fire Plant Response Model. The Internal Fire Plant Response Model verification typically includes the following:

(a) the fire-induced initiating events are properly identified;

(b) the equipment (e.g., SSCs, instrumentation, barriers) is properly modeled with the appropriate fire relevant failure modes, including spurious operation and accounting for the appropriate fire scenarios;

(c) the modeled equipment and HFEs represent the as-built, as operated plant considering the reactor type, design vintage, and specific design;

(d) the HFEs are properly modeled, including both non-fire-specific and fire relevant actions.;

(e) Findings associated with the internal events analysis have been dispositioned such that they do not adversely impact the fire PRA.

4.3.9.11.3.6 Internal Fire Scenario Selection and Analysis

A review shall be performed on the Internal Fire Scenario Selection and Analysis process. The Internal Fire Scenario Selection and Analysis process verification typically includes the following:

(a) the Internal Fire Scenario Selection and Analysis technical subelement has identified and analyzed a representative set of fire scenarios that adequately cover potential scenarios involving fire that are risk-significant contributors for both single and multi-compartment scenarios as appropriate;

(b) the selected target sets are reasonable and appropriately represent potential post-fire cable and equipment failures, including specification of failure modes, such as spurious operations, given the nature of the fire sources present and target locations;

(c) fire detection and suppression considerations have been analyzed appropriately;

(d) appropriate fire modeling tools have been selected, and the fire modeling tools have been applied within their capabilities and limitations by personnel knowledgeable of their use.

4.3.9.11.3.7 Internal Fire Ignition Frequency

A review shall be performed on the Internal Fire Ignition Frequency analysis. The Internal Fire Ignition Frequency analysis verification typically includes the following:

(a) the ignition frequencies have included generic industry data and experience;

(b) as appropriate to the Capability Category, the ignition frequency analysis has considered plant outlier experience and/or has included plant-specific frequency updates;

(c) the apportionment process applied to estimate fire area, fire compartment, and/or fire scenario frequencies has appropriately preserved the original plant-wide fire frequencies for all ignition sources.

4.3.9.11.3.8 Internal Fire Circuit Failure Analysis

A review shall be performed on the Internal Fire Circuit Failure Analysis. The Internal Fire Circuit Failure Analysis verification typically includes the following:

(a) for a selected set of representative cases, the circuit failure analysis has appropriately identified the relevant fire-induced circuit failure modes;

(b) for a selected set of representative cases, the circuit failure mode probability evaluations have appropriately quantified the likelihood of fire-related failure modes that could cause equipment functional failure and/or spurious operation.

4.3.9.11.3.9 Internal Fire Human Reliability Analysis

A review shall be performed on the Internal Fire Human Reliability Analysis. The Internal Fire Human Reliability Analysis verification typically includes the following:

(a) the Internal Fire Human Reliability Analysis adequately includes the additional influences caused by fire;

(b) HFEs adopted from an internal events PRA have been modified as appropriate to represent fire effects;

(c) new HFEs are included to incorporate specific fire-related actions that are consistent with plant procedures that were not covered by the internal events PRA.

4.3.9.11.3.10 Internal Fire Event Sequence Quantification

A review shall be performed on the Internal Fire Event Sequence Quantification. The Internal Fire Event Sequence Quantification verification typically includes the following:

(a) the event sequence and event sequence family for each quantified fire scenario is properly quantified;

(b) the internal fire PRA provides the results and insights needed for risk-informed decisions;

(c) the event sequence and event sequence family estimates and uncertainties have been reported;

(d) the risk-significant contributors have been identified and discussed.

4.3.9.12 References for Internal Fire PRA

The following is a list of publications referenced in this Standard.

[F-1] EPRI TR-1011989-NUREG/CR-6850: EPRI/NRC-RES, "Fire PRA Methodology for Nuclear Power Facilities," NRC has endorsed the original 2001 version of NFPA-805 but not the 2006 revision. A joint EPRI/NRC publication, EPRI TR-1011989, Palo Alto, CA, and NUREG/CR-6850, U.S. NRC, Washington, DC, (a report in two volumes); Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852, September 2005

[F-2] NEI 00-01, (Rev. 3) "Guidance for Post Fire Safe Shutdown Circuit Analysis," 2011

[F-3] NFPA Standard 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," Publisher: National Fire Protection Association (NFPA), 1 Batterymarch Park, Quincy, MA 02169, 2001

[F-4] NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," Final Report (NUREG/CR-7150, Volume 1), October 2012

[F-5] NUREG-2169, EPRI 3002002936, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database: United States Fire Event Experience Through 2009," January 2015

[F-6] ASTM Standard E119 10b, "Standard Test Methods for Fire Tests of Building Construction and Materials," Publisher: American Society for Testing and Materials (ASTM International), 100 Barr Harbor Drive, P.O. Box C700, West Conshohocken, PA 19428, October 2010

[F-7] NUREG 1921, EPRI/NRC-RES, "Fire Human Reliability Analysis Guidelines," May 2012

[F-8] Information Notice No. 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire," February 28, 1992

[F-9] NFPA Standard 520, "Standard on Subterranean Spaces," Publisher: National Fire Protection Association (NFPA), 1 Batterymarch Park, Quincy, MA 02169, 2010

NONMANDATORY APPENDIX F: NOTES AND EXPLANATORY MATERIAL FOR INTERNAL FIRE PRA

F.1 NOTES ASSOCIATED WITH INTERNAL FIRE PRA

Table F-1 Notes Supporting Internal Fire PRA Requirements

Number	Notes
F-N-1	The intent of this requirement is that the global analysis boundary will include locations that may contain fire sources that could threaten equipment included in the Internal Fire Plant Response Model or related cables by virtue of a multi-compartment fire scenario but that may not themselves contain included equipment or cable items. See FPP-A1
F-N-2	The intent of this requirement is to include sister reactor locations that meet the selection criteria as stated. See FPP-A1
F-N-3	The intent of Requirement FPP-B2 is to allow an analysis to credit partitioning features that have a specific fire-endurance rating in the plant partitioning analysis without further justification subject only to the restriction imposed by Requirement FPP-B4. However, plant partitioning may also, with justification, credit partitioning features that lack a specific fire endurance rating (nonrated elements) such as spatial separation or nonrated structural elements. See FPP-B2
F-N-4	Volume 2, Chapter 1, of NUREG/CR-6850 [F-1] discusses criteria that may be applied in justifying decisions related to spatial separation, active fire barrier elements, and partitioning features that lack a fire-resistance rating. See FPP-B2
F-N-5	Examples of investigations include, but are not limited to, actives such as tabletop review(s), physical/computerized walkdown(s) of the plant, and/or interview(s) with knowledgeable personnel. These investigation(s) may need to be supported by a review of plant documents, especially in the context of external hazard evaluations. This SR does not imply that each of these activities need to be performed to meet the SR. Further, if appropriate, one investigation can meet the intent of multiple investigation SRs if the investigation is scoped and structured appropriately. See FPP-B5, FSS-D9, FSS-D10
F-N-6	This SR is likely not applicable to PRAs performed during the pre-operational stage. See FPP-B5, FPRM-B3, FSS-D10, FIGN-A5, FHR-A2
F-N-7	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See FPP-B8, FES-C3, FQLS-A6, FPRM-B17, FSS-G9, FIGN-A12, FCF-A4, FHR-D3, FESQ-D3
F-N-8	This SR is not applicable to operating plants. See FPP-B8, FPP-C3, FES-C3, FES-D3, FCS-B3, FCS-C3, FQLS-A6, FQLS-B3, FPRM-B4, FPRM-B17, FPRM-C4, FSS-D11, FSS-E2, FSS-G9, FSS-H4, FIGN-A4, FIGN-A6, FIGN-A12, FIGN-B3, FCF-A4, FCF-B3, FHR-A3, FHR-D3, FHR-E3, FESQ-D3, FESQ-F3

Table F-1 Notes Supporting Internal Fire PRA Requirements (Cont'd)

Number	Notes
F-N-9	In the context of Requirement FES-A2, it is acceptable to define "equipment" as the system whose failure causes the initiating event. That is, it is not intended that this requirement require that every individual piece of equipment throughout the plant whose failure might lead to an initiating event be identified explicitly. Rather, "equipment" might be identified at a higher level (e.g., at the system-level). For example, the internal fire PRA may not choose to treat a certain balance of plant (BOP) systems at the same level of detail as are other plant systems. This decision may, for example, be driven by a lack of cable routing information for the system in question. In such a case, the BOP system might be treated as, in effect, a "supercomponent" whose failure would lead to an initiating event. Such approaches are intended to be acceptable so long as the uncertainty introduced by such assumptions is acceptable under the requirements of Internal Fire Event Sequence Quantification. See FES-A2
F-N-10	It is understood that equipment extends to the specific piece of equipment itself and any supportive equipment (e.g., power supply, associated actuating instrumentation, and interlocks) needed to perform the intended operation/function of the primary equipment item. In recognition that it is impractical to explicitly identify and locate all equipment and their cables that could contribute to or cause an initiating event such as, for instance, all the BOP equipment, the intent of this requirement is to allow the analyst to use other levels of equipment definition (e.g., rather than identifying and locating individual equipment items in the secondary coolant system such as the pumps and valves, the analyst chooses to identify the equipment more globally as "secondary coolant system"). This can be done as long as its failure in terms of an initiating event is analyzed conservatively (for instance, analyzing any failure of the secondary coolant system as causing an unrecoverable total loss of cooling/heat sink initiating event even though some individual equipment item failures may not actually cause a total unrecoverable loss of the entire system). See FES-A2
F-N-11	The NEI 00-01 [F-2] process for identifying multiple spurious operation (MSO) combinations for deterministic safe shutdown analysis would be one acceptable method for meeting Requirement FES-A3 if that process is extended to include PRA systems and functions not included in the scope of the safe shutdown analysis. In some regards, the NEI 00-01 process actually exceeds the scope of analysis specified in companion Requirements FES-A4, FES-A5, and FES-A6. In other regards, it may be incomplete relative to internal fire PRA in that some internal fire PRA systems will likely have been excluded from the scope of an NEI 00-01 analysis. See FES-A3
F-N-12	For plants adopting NFPA 805 [F-3], the Nuclear Safety Capability Assessment is used in lieu of Fire Safe Shutdown/Appendix R Analysis in the context of Requirements FES-A4, FES-A5, and FES-A6. See FES-A4, FES-A5, FES-A6
F-N-13	Fire-induced failures leading to a loss of heat sink or radionuclide transport barrier bypass are examples of cases where fire-induced failures could contribute to initiating events, event sequences, and event sequence families. See FES-A6
F-N-14	Random failures do not need to be included in the analyses for this requirement. See FES-A6
F-N-15	This requirement also addresses part of the SRs for HLR-FES-B by addressing the operability/functionality of portions of the plant design that may be credited in the internal fire PRA. See FES-A6

Table F-1 Notes Supporting Internal Fire PRA Requirements (Cont'd)

Number	Notes
F-N-16	Exclusion of equipment or failure modes such as MSOs during the equipment selection phase can be performed given sufficient justification. For example, NUREG/CR-7150 [F-4] identifies certain cable failure modes (or combinations) that are considered incredible, and these modes (or combinations) could be excluded from the internal fire PRA on that basis. See FES-A7, FES-B3
F-N-17	The NEI 00-01 [F-2] process for identifying MSO combinations for deterministic safe shutdown analysis would be one acceptable method for meeting Requirement FES-B2 if that process is extended to include PRA systems and functions not included in the scope of the safe shutdown analysis. In some regards, the NEI 00-01 process actually exceeds the scope of analysis specified in companion Requirement FES-B3. In other regards, it may be incomplete relative to internal fire PRA in that some internal fire PRA systems will likely have been excluded from the scope of a NEI 00-01 analysis. See FES-B2
F-N-18	Instrumentation needs to be included because of the higher probability for fire-induced indication failure including spurious indications as compared to the potential for random indication failure. Hence, while random failures of instrumentation may often be ignored in an internal events PRA, fire-induced instrumentation failure needs to be included in an internal fire PRA. Inclusion of just one fire-induced spurious indication relevant to each operator action being addressed for CC-II is indicative of balancing: *(a)* the current state of the art and the resources required to consider almost innumerable combinations of two or more spurious indications against; *(b)* the desire to include in the internal fire PRA the associated risk caused by such spurious indications. See FES-C1
F-N-19	Documentation does not necessarily imply a separate/unique list of equipment, although this may prove useful. For instance, inclusion in the Internal Fire Plant Response Model can be a part of “documenting” the equipment included and its failure modes. The ability to create such a list should exist especially for peer review efficiency as well as for conducting the internal fire PRA itself. See FES-D1
F-N-20	NEI-00-01 [F-2] Chapter 3 provides one acceptable method for performing circuit failure analysis for circuits identified in the Internal Fire PRA. See FCS-A1
F-N-21	The distinction between CC-I and CC-II is on the resolution of cable mapping to failure modes (i.e. basic event). In CC-II, specific cables are mapped to the appropriate failure mode based on the circuit analysis for equipment that are risk-significant contributors. See FCS-A1
F-N-22	In the context of Requirement FES-A2 (see explanatory note for Requirement FES-A2), one acceptable approach to the identification of internal fire PRA equipment is to define “equipment” as the system whose failure causes the initiating event rather than identifying every individual piece of equipment throughout the plant whose failure might lead to an initiating event. In the context of Requirement FCS-A2, where the “supercomponent” approach has been applied, a similar approach to cable selection and location would be expected. That is, in such cases, it is acceptable that individual cables supporting the “supercomponent” system might not be identified and routed in detail, but rather, simply be associated as a group to various plant locations such that the failure of any support cable would cause failure of the supercomponent/system. See FCS-A2

Table F-1 Notes Supporting Internal Fire PRA Requirements (Cont'd)

Number	Notes
F-N-23	The intent of Requirement FCS-A2 is, in part, to provide limits on the scope of instruments to be identified in accordance with the risk significance of operator actions included in the Internal Fire Plant Response Model. For example, if the use of a conservative screening HEP shows that an operator action is not a risk-significant contributor, then the analyst may choose not to identify instrumentation and, by implication of Requirement FCS-A1, not to complete cable tracing for such instruments. However, it is intended that this requirement will require that the instruments that are relied on for included operator actions will be identified and verified as available to a level of detail commensurate with the risk significance and for quantification of the HEPs. See FCS-A2
F-N-24	Exact routing information is expected to be known for operating reactors. See FCS-A3, FQLS-A1
F-N-25	A cable terminal end location refers to the location where each end of the cable is terminated at some piece of plant equipment. In some cases, the cable might enter this equipment from the floor below. In these cases, the cable routing information must incorporate the presence of the cable in the fire area or fire compartment where it is actually terminated. See FCS-A4
F-N-26	The internal fire PRA may make conservative assumptions regarding cable locations. That is, if the exact routing of a cable (or group of cables) has not been established, the internal fire PRA should assume that those cables fail for any fire scenario that has a damaging effect on any raceway or location where the subject cable might reasonably exist. The resulting Capability Category if this option is taken is to be based on the general guidance stated in Table 1.4-2 for both resolution and realism. The determination of where cables might reasonably exist should include the physical layout of the plant equipment and the routing of cables analyzed explicitly using Requirement FCS-A3 from nearby or identical locations. For PRAs performed prior to construction, cable location assumptions should be limited to general, less specific locations since detailed plant spatial features are generally not known prior to construction. The intent is to allow for the application of conservative assumptions in cases where the specific routing of a cable is not known. See FCS-A4
F-N-27	The internal fire PRA should strive for completeness in its cable routing information. It is acknowledged, however, that practicality may limit the completeness of cable routing information. If full cable routing information is not developed, the routing of cables on an exclusionary basis is acceptable. That is, if it can be established (based on the physical features and layout of the plant) that a particular cable (or group of cables) is not routed through a given PAU (or specific location within a PAU), then the internal fire PRA may assume that the excluded cable(s) will not fail for fire scenarios where fire-induced damage is limited to that PAU (or to a specific location within a PAU). See FCS-A4
F-N-28	It is acceptable for the qualitative screening analysis to retain any PAU for quantitative analysis without a rigorous application of the defined qualitative screening criteria. See FQLS-A1, FQLS-A2, FQLS-A3
F-N-29	Internal fire PRA practice may involve screening out PAUs if the time available before a required shutdown due to a Technical Specification violation is long. This Section does not establish a specific time limit but acknowledges the potential validity of this approach. It is expected that analysts will define and provide a basis for their approach if an upper-bound time limit is applied beyond which a shutdown required by the Technical Specifications will not be included as an initiating event. See FQLS-A2
F-N-30	Requirements FQLS-A1, FQLS-A2, and FQLS-A3 represent minimum criteria. The intent of Requirement FQLS-A4 is to allow for the application of additional screening criteria. However, if additional criteria are applied, then they must be defined, and a basis for their acceptability must be established. See FQLS-A4

Table F-1 Notes Supporting Internal Fire PRA Requirements (Cont'd)

Number	Notes
F-N-31	The internal events PRA model may or may not include event sequences involving two or more reactors or radionuclide sources. However, if an internal fire and its propagation path impacts two or more reactors or sources, such event sequences may be introduced for treatment of this hazard group. See FPRM-B1
F-N-32	In internal fire PRA practice, multiple ignition sources can be analyzed by using a single fire scenario (e.g., a bank of several similar electrical panels might be grouped and analyzed with a single fire scenario), provided that the assumed fire ignition frequency and fire characteristics bound the cumulative contribution of all of the individual ignition sources included under the selected fire scenario. See FSS-A4
F-N-33	It is expected that the number of individual fire scenarios and the level of detail included in the analysis of each scenario will be commensurate with the risk significance, for fire, of the PAU under analysis. PAUs with small risk contribution may, for example, be characterized based on the conservative analysis of a single bounding fire scenario. The more risk-significant PAUs will likely be characterized by detailed analysis of multiple and/or more specific fire scenarios. In particular, those PAUs that are identified as the risk-significant fire risk contributors should be characterized by the detailed quantification (see the SRs of HLR-FSS-C) of one or more fire scenarios that combine specific ignition sources and specific target sets. See FSS-A4
F-N-34	Conditions and assumptions that could influence whether or not a fire will damage targets include, for example, the distance between fire source and target, position of the targets relative to the fire source, the damage threshold of the targets, and the mode of fire exposure (e.g., buoyant plume exposure versus radiant heating). See FSS-C4
F-N-35	SRs in HLR-FSS-G provide for the analysis of fire scenarios impacting adjacent PAUs (the multi-compartment fire analysis). Requirement FSS-C7 is intended, in part, to ensure that a similar analysis is included for cases where fire barriers exist within a single PAU (i.e., the fire barriers exist but were not credited during plant partitioning). If the analysis of fire scenarios within a single PAU credits these fire barriers (e.g., with limiting fire damage, or delaying the spread of fire or the onset of fire damage), then Requirement FSS-C7 requires an analysis of fire scenarios involving the failure of the credited fire barrier that is analogous to the multi-compartment fire analysis. Such fire barriers may include passive fire barriers (e.g., nonrated partition walls, cable wraps, or radiant energy shields) or active fire barriers (e.g., normally open fire doors or water curtains). See FSS-C7
F-N-36	The National Fire Protection Association (NFPA) standard 520 (2010 version) [F-9] defines “high hazard” fire sources as “contents that are likely to burn with extreme rapidity or from which explosions are likely.” In the context of a nuclear power plant, this would equate to the presence or potential release of large quantities of flammable liquid or hydrogen gas. See FSS-C7
F-N-37	The intent of this requirement is to cover parameters used in fire modeling that are not explicitly associated with characterizing the fire scenario configuration. Examples of these parameters may include thermo-physical properties of boundary materials, ambient temperature, etc. See FSS-D3
F-N-38	An example of a statistical fire model would be one where fire spread behavior within electrical panels or the main control board has been modeled statistically. A second example might be the modeling of fire intensity using a probability distribution. See FSS-D6

Table F-1 Notes Supporting Internal Fire PRA Requirements (Cont'd)

Number	Notes
F-N-39	An empirical model, as that term is used here, is a fire model based on experience or observation alone. For example, fire suppression by the manual fire brigade is often based on an empirical relationship derived from a statistical analysis of fire suppression times reported in past operating experience. A second example is characterizing high-energy arcing faults in electrical switching equipment based on characteristics observed in past events. A third example is the wide range of closed-form empirical correlations documented in sources such as textbooks or engineering handbooks. See FSS-D7
F-N-40	Fire scenarios that assume widespread damage (e.g., damage across an entire PAU) will generally include potential smoke damage within the limits of the assumed fire damage (e.g., assuming the loss of all equipment in a PAU given a fire, as might be employed during the early stages of a screening analysis). See FSS-D8
F-N-41	A screening level fire scenario analysis that assumes widespread fire damage within a PAU would only require confirmation of sources and targets present in the PAU. See FSS-D9, FSS-D10, FSS-D11
F-N-42	The applicable codes and standards will generally be the relevant NFPA code(s) of record. See FSS-E1
F-N-43	The total system unavailability is intended to represent functional performance of the system (e.g., a detector system may function even though one or more individual detectors are out of service or fail). Also note that total system unavailability includes unreliability and unavailability. See FSS-E1
F-N-44	Typical internal fire PRA practice involves the application of a nonsuppression probability; that is, the probability that suppression efforts fail to suppress the fire before the onset of the postulated equipment/cable damage. Hence, the nonsuppression probability estimate includes an assessment of effectiveness (including the relative timing of fire damage versus detection/suppression and fire brigade performance), discussed in Requirement FSS-E3, as well as an overall assessment of system unavailability. The intent of Requirement FSS-E1 is to require increasing levels of plant-specificity in assessing system unavailability with increasing Capability Category. See FSS-E1
F-N-45	The intent for CC-II is to additionally require a review of plant records to determine whether the generic unavailability estimate is consistent with actual system unavailability. Outlier experience would be any experience indicating that actual system is unavailable more frequently than would be indicated by the generic values. See FSS-E1
F-N-46	This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding. See FSS-E4, FIGN-A10, FCF-A2
F-N-47	The statistical treatment of manual fire suppression is typically complementary to the events included when fire frequency is estimated and, as a result, the two factors are typically highly dependent. For example, in at least one early fire frequency analysis, any fire that lasted less than five minutes was eliminated from the fire frequency event pool as non-challenging to plant safety. If these frequency values were used in an internal fire PRA, it would be necessary to assume that all of the fire scenarios analyzed in the PRA last a minimum of five minutes prior to suppression to maintain consistency. If the analysis were to instead credit early fire suppression by general plant personnel (i.e., suppression in less than five minutes) then this would be an example of “double counting” the fire suppression credit. See FSS-E5

Table F-1 Notes Supporting Internal Fire PRA Requirements (Cont'd)

Number	Notes
F-N-48	The use of the ignition frequency and fire suppression values published in EPRI 3002002936 [F-5] is one acceptable method to meet this SR. That is, the analyses performed in accordance with the EPRI report meet the requirements of this SR. See FSS-E5
F-N-49	The prototypical fire scenario leading to failure of structural steel would be catastrophic failure of the turbine itself (e.g., a blade ejection event) and an ensuing lube-oil fire. For the lube-oil fire, the possibility of effects of pooling, the flaming oil traversing multiple levels, and spraying from continued lube-oil pump operation should be included. Additional examples would include scenarios involving other high-hazard fire sources as present in the relevant PAUs (e.g., oil storage tanks, hydrogen storage tanks and piping, mineral oil-filled transformers). See FSS-F1
F-N-50	In applying Requirements FSS-C1, FSS-C2, FSS-C3, FSS-C4, FSS-C5, FSS-C6, and FSS-C7, additional phenomena associated with multi-compartment fire scenarios, beyond those associated with scenarios of single PAUs, may be addressed. For example, the modeling of hot gas flow through openings and ducts from the PAU of fire origin may be necessary. See FSS-G1
F-N-51	Passive fire barrier features that may have been credited in plant partitioning or scenario analysis include items such as walls, normally closed fire doors, penetration seals, and other similar features that require no action (manual or automatic) to perform their intended function. This requirement would apply to all passive fire barrier elements credited in the fire PRA, including the plant partitioning, as well as in the fire-scenario selection and analysis. The fire-resistance rating of passive fire barrier features is typically established in accordance with the ASTM E 119-10b [F-6] test standard and/or other similar, related, or subsidiary standards. See FSS-G4
F-N-52	Active fire barrier elements include items such as normally open fire doors, dampers, water curtains, and other similar items that require that some action (manual or automatic) occur for the element to perform its intended function. See FSS-G6
F-N-53	This SR is not applicable if similar plants do not exist (i.e., if the plant is a first-of-a-kind design). See FIGN-A1
F-N-54	Outlier experience includes cases where the plant has experienced more fires of any given type than would be expected given the generic industry experience, or where the plant has experienced a type of fire that is potentially risk relevant but is not represented in the generic event database. See FIGN-A4
F-N-55	That is, the analysis addresses the fraction of the year that the plant is in at-power operational state. For further discussion see the explanatory note associated with Requirement IE-C7. See FIGN-A5, FIGN-A6
F-N-56	This SR is only relevant when combining generic and plant-specific data. See FIGN-A5
F-N-57	Technology-specific testing evidence includes testing data collected from relevant nuclear and non-nuclear testing programs for the component or system of interest. An example includes evidence collected from a separate effects test loop that will employ the same valve under approximately the same temperatures and working fluid conditions as modeled in the PRA. See FIGN-A7
F-N-58	An example of a “plant-wide consistent methodology” would be one where if equipment count is chosen as the approach for determining PAU apportioning factors, counting rules should be established and applied consistently throughout all the PAUs in the plant and that preserves the plant-wide fire frequency. See FIGN-A8

Table F-1 Notes Supporting Internal Fire PRA Requirements (Cont'd)

Number	Notes
F-N-59	The analysis must include all potential ignitions sources, both fixed and transient. See FIGN-A9
F-N-60	Acceptable methods include Bayesian updating or expert judgment consistent with Section 4.2 of this Standard. See FIGN-A10
F-N-61	The requirements under Internal Fire Plant Response Model address HFEs carried over from the internal events analysis. Requirement FHR-A1 addresses new HFEs that are unique to the fire analysis. See FHR-A1
F-N-62	HFEs related to actions previously modeled in an analysis such as the internal events PRA may have to be modified because the fire may change the scenario characteristics such as timing, cues, or specific actions that would have to be taken (e.g., due to fire-induced circuit failures that affect the manner in which certain components may be operated). These changes would therefore require alteration of a previously defined HFE to fit the applicable fire situation in the internal fire PRA. See FHR-B2
F-N-63	One acceptable method for meeting this requirement is stated in NUREG-1921 [F-7], including its definition of detailed analysis versus screening/scoping methods. See FHR-C1
F-N-64	An example of a fire-related effect that must be analyzed carefully in identifying and evaluating recovery actions is the potential for a circuit failure that could both defeat automatic operation of a valve and prevent remote manual operation (see Information Notice 92-18, [F-8]). See FHR-D1
F-N-65	In some cases, a given fire scenario could lead to more than one initiating event. For screening purposes, the selection of the most conservative initiating event might be assumed with a conditional probability of 1.0 for the corresponding pump failure mode. Quantification might also consider both initiators with a split fraction applied to incorporate each pump failure mode. The intent of Requirement FESQ-A2 is to ensure that the selected initiating event, or events, encompasses the risk contribution from all applicable initiating events. See FESQ-A2
F-N-66	When quantifying fire scenarios based on an internal events initiating event sequence, there may be a difference in success criteria, timing of human actions, and other elements of the PRA model for a fire-induced system failure that causes a demand for a reactor trip and the same failure if it occurs after a reactor trip. If, for example, the internal fire PRA model employs a general transient as the initiating event, with all of the fire impacts included as failures subsequent to that trip, then to meet the intent of Requirement FESQ-A2, it would be appropriate to ensure that any differences with respect to selecting a more specific initiating event are negligible. See FESQ-A2
F-N-67	There is no requirement for a comparison of internal fire PRA results for similar plants under this SR, due to lack of publicly available internal fire PRA results. Additionally, small differences in geometry, plant layout, and the fire safe shutdown procedures may result in large differences in risk that may be difficult to understand without detailed internal fire PRA results from plants being compared. See FESQ-D1

4.3.10 Seismic PRA (S)

This Section presents the technical requirements associated with Seismic PRA.

The requirements in this Section are divided into the following technical subelements:

(a) Seismic Hazard Analysis (SHA);
(b) Seismic Fragility Analysis (SFR);
(c) Seismic Plant Response Model (SPR).

4.3.10.1 Objectives and Technical Requirements for Seismic Hazard Analysis (SHA)

The objectives of the Seismic Hazard Analysis are as follows:

(a) identify a site or range of sites and provide the basis for exceedance frequencies;
(b) characterize uncertainties associated with the site-specific hazard evaluation;
(c) assess the frequency of occurrence of seismic ground motions in the region associated with the site or range of site;
(d) determine the range of seismic vibratory ground motion associated with earthquakes at given locations, types, and magnitudes;
(e) include the effects of local site response;
(f) propagate uncertainties;
(g) incorporate the results of the hazard analysis when determining site-specific spectral shapes for Seismic Fragility Analysis;
(h) assess whether other seismic hazards need to be included;
(i) the Seismic Hazard Analysis is documented to provide traceability of the work.

Table 4.3.10.1-1 High Level Requirements for Seismic Hazard Analysis

Designator	Requirement
HLR-SHA-A	The basis for the calculation of the frequencies of exceeding different levels of horizontal vibratory seismic ground motion at the site shall be a probabilistic seismic hazard analysis specific to a site that represents the center, body, and range of the technically defensible interpretations.
HLR-SHA-B	Inputs to the Seismic Hazard Analysis shall include characterization of uncertainty and shall be based on current geological, seismological, and geophysical data; local site topography; and surficial geologic and geotechnical site properties. A catalog of historical, instrumental, and paleoseismicity information shall be compiled. Available models and methods shall be compiled.
HLR-SHA-C	To assess the frequency of exceedance of seismic ground motions in the site region, the probabilistic seismic hazard analysis shall examine all credible sources of potentially damaging earthquakes. Uncertainties in the seismic source characterization shall be identified and addressed.
HLR-SHA-D	The probabilistic seismic hazard analysis shall include a ground motion characterization model that determines the range of seismic horizontal vibratory ground motion that can occur at a site given the occurrence of an earthquake at a specific location and of a certain type (e.g., strike slip, normal, reverse) and magnitude. Uncertainties in characterizing the ground motion propagation shall be identified and included.
HLR-SHA-E	The Seismic Hazard Analysis shall include the effects of local site response. Uncertainties in characterizing the local site response analysis shall be identified and included.
HLR-SHA-F	Aleatory and epistemic uncertainty in each step of the hazard analysis shall be propagated in the final quantification of hazard estimates for the site.
HLR-SHA-G	For further use in the Seismic Fragility Analysis, the spectral shape shall be based on-site-specific evaluation that considers or incorporates the results of the hazard evaluation.
HLR-SHA-H	An evaluation shall be performed to assess whether, in addition to the vibratory ground motion, other seismic hazards need to be included in the Seismic PRA.
HLR-SHA-I	The documentation of the Seismic Hazard Analysis shall provide traceability of the work.

Table 4.3.10.1-2 Supporting Requirements for HLR-SHA-A

The basis for the calculation of the frequencies of exceeding different levels of horizontal vibratory seismic ground motion at the site shall be a probabilistic seismic hazard analysis specific to a site that represents the center, body, and range of the technically defensible interpretations. (HLR-SHA-A)

Index No. SHA-A	Capability Category I	Capability Category II
SHA-A1	For the Seismic Hazard Analysis, either: *(a)* IDENTIFY the site at which the reactor being analyzed is located; or *(b)* DESCRIBE a bounding site and JUSTIFY that the bounding site bounds the list of sites in the scope of the PRA. See Note S-N-1	
SHA-A2	USE a defined process to develop the probabilistic seismic hazard analysis model to ensure that the probabilistic seismic hazard analysis represents the center, body, and range of the technically defensible interpretations. See Note S-N-2	
SHA-A3	USE either the spectral accelerations, the average spectral acceleration over a selected band of frequencies, or the peak ground acceleration as the parameter to characterize both hazard and fragilities.	
SHA-A4	ENSURE that the ground motion parameter(s) and range of frequencies selected for the probabilistic seismic hazard analysis are consistent with the ground motion parameter(s) needed for subsequent fragility and plant response analysis. See Requirement SPR-E1	
SHA-A5	In developing the probabilistic seismic hazard analysis results for use in Event Sequence Quantification (whether they are characterized by spectral accelerations, peak ground accelerations, or both), EXTEND the range of ground motions considered to large enough values (consistent with the available earth science and interpretations), such that the truncation does not affect final numerical results (e.g., on parameters such as event sequence family frequencies) and the delineation and ranking of seismic-induced sequences are not affected.	
SHA-A6	JUSTIFY the specified lower bound magnitude for use in the hazard analysis, such that earthquakes of magnitudes less than this value are not expected to cause damage to the engineered structures or equipment. See Note S-N-3	
SHA-A7	JUSTIFY the specified number of standard deviations (epsilon) from the median of the lognormal distribution of the ground motion value (e.g., spectral acceleration) to be included in the analysis of the ground motion prediction equation such that aleatory variability in the ground motion prediction is properly modeled. See Note S-N-4	

Table 4.3.10.1-3 Supporting Requirements for HLR-SHA-B

Inputs to the Seismic Hazard Analysis shall include characterization of uncertainty and shall be based on current geological, seismological, and geophysical data; local site topography; and surficial geologic and geotechnical site properties. A catalog of historical, instrumental, and paleoseismicity information shall be compiled. Available models and methods shall be compiled. (HLR-SHA-B)

Index No. SHA-B	Capability Category I	Capability Category II
SHA-B1	In performing the Seismic Hazard Analysis, USE current geological, seismological, geophysical, and geotechnical data that are used by subject matter experts/analysts to develop interpretations and inputs to the probabilistic seismic hazard analysis. See Note S-N-5	
SHA-B2	ENSURE that the size of the region to be investigated and the data and information used are adequate to characterize all credible seismic sources that may be a major contributor to the seismic hazard at the site and the uncertainties associated with the hazard results.	
SHA-B3	ENSURE that the data and information are sufficient to characterize attributes important for modeling both regional propagation of ground motions and local site effects including their associated uncertainties.	
SHA-B4	ENSURE that new data, models, methods, and interpretations unknown when the existing models were developed or not previously used, and that could affect an existing probabilistic seismic hazard analysis, are identified and compiled. See Note S-N-6	
SHA-B5	USE a compiled catalog of historically reported earthquakes, instrumentally recorded earthquakes, and earthquakes identified through geological investigations in performing the probabilistic seismic hazard analysis.	

Table 4.3.10.1-4 Supporting Requirements for HLR-SHA-C

To assess the frequency of occurrence of seismic ground motions in the site region, the probabilistic seismic hazard analysis shall examine all credible sources of potentially damaging earthquakes. Uncertainties in the seismic source characterization shall be identified and addressed. (HLR-SHA-C)

Index No. SHA-C	Capability Category I	Capability Category II
SHA-C1	In the probabilistic seismic hazard analysis, IDENTIFY the sources of earthquakes that have the potential to be a major contributor to the seismic hazard at the site.	
SHA-C2	USE a structured approach to characterize seismic sources using the information compiled in accordance with the Supporting Requirements (SRs) of HLR-SHA-A and HLR-SHA-B. See Note S-N-7	
SHA-C3	USE a structured approach to identify and include sources of uncertainty in the modeling of the seismic sources. See Note S-N-8	
SHA-C4	If an existing seismic source model is used, DEMONSTRATE that new seismic sources, models, and methods unknown when the existing models were developed or not previously used are included in the center, body, and range of the existing model and do not challenge the technical validity of the existing model, or include them using an appropriate level of analysis in an update of the model. See SHA-C2	
SHA-C5	If an existing seismic source model is updated for use in the probabilistic seismic hazard analysis, JUSTIFY the level and method of analysis used in the update of the model.	

Table 4.3.10.1-5 Supporting Requirements for HLR-SHA-D

The probabilistic seismic hazard analysis shall include a ground motion characterization model that determines the range of seismic horizontal vibratory ground motion that can occur at a site given the occurrence of an earthquake at a specific location and of a certain type (e.g., strike slip, normal, reverse) and magnitude. Uncertainties in characterizing the ground motion propagation shall be identified and included. (HLR-SHA-D)

Index No. SHA-D	Capability Category I	Capability Category II
SHA-D1	The ground motion characterization model that determines the range of seismic vibratory ground motion that can occur at a site INCLUDES *(a)* credible mechanisms governing estimates of vibratory ground motion that can occur at a site; *(b)* a review of available historical and instrumental seismicity data (including strong motion data) to assess and calibrate the model; *(c)* criteria for selection of (existing and/or newly developed) ground motion prediction equations for the ground motion estimates; and *(d)* reference soil or rock horizon (defined by shear wave velocity, density, and damping values).	
SHA-D2	ENSURE that the process used to characterize the ground motion or the other elements of the ground motion analysis is compatible with the level of analysis discussed in the SRs of HLR-SHA-A. See Note S-N-7	
SHA-D3	ENSURE that uncertainties are included in the model such that the aggregate of predicted ground motions captures the range of ground motions that can occur at a site in accordance with the level of analysis identified for the SRs of HLR-SHA-A and the data and information identified in the SRs of HLR-SHA-B. For seismic events analyses that are intended to cover a range of sites, INCLUDE the site-to-site variability in the assessment of uncertainty, or JUSTIFY the selection of a site that bounds the applicable site-to-site variability.	
SHA-D4	If existing ground motion models are used, DEMONSTRATE that available new data, models, methods, and information (unknown when the existing models were developed or not previously used) would not affect the probabilistic seismic hazard analysis ground motion results or INCLUDE new ground motion data, models, methods, and information in the update of the probabilistic seismic hazard analysis. See SHA-D2	

Table 4.3.10.1-6 Supporting Requirements for HLR-SHA-E

The Seismic Hazard Analysis shall include the effects of local site response. Uncertainties in characterizing the local site response analysis shall be identified and included. (HLR-SHA-E)

Index No. SHA-E	Capability Category I	Capability Category II
SHA-E1	In the Seismic Hazard Analysis, INCLUDE the effects of site topography, surficial geologic deposits, and site geotechnical properties on ground motions at the site. Meet Requirement SHA-E2, if applicable.	
SHA-E2	For PRAs performed using a bounding site, INCLUDE the site-to-site variability when satisfying Requirement SHA-E1. See Note S-N-9	
SHA-E3	INCLUDE uncertainties in the local site response analysis. Meet Requirement SHA-E4, if applicable.	
SHA-E4	For PRAs performed using a bounding site, INCLUDE the site-to-site variability when satisfying SHA-E3. See Note S-N-9	
SHA-E5	JUSTIFY the approach used to incorporate the site response analysis into the hazard analysis (e.g., sources of soils and rock material properties used in the analysis, uncertainties in site characterization and material properties, data to identify the depth to bedrock, appropriateness of one-, two-, or three-dimensional analysis in relation to the site stratigraphy). Meet Requirement SHA-E6, if applicable.	
SHA-E6	For PRAs performed using a bounding site, INCLUDE the site-to-site variability when satisfying Requirement SHA-E5. See Note S-N-9	

Table 4.3.10.1-7 Supporting Requirements for HLR-SHA-F

Aleatory and epistemic uncertainty in each step of the hazard analysis shall be propagated in the final quantification of hazard estimates for the site. (HLR-SHA-F)

Index No. SHA-F	Capability Category I	Capability Category II
SHA-F1	CALCULATE the following results as a part of the hazard quantification process, compatible with the level of analysis determined in the SRs of HLR-SHA-A: *(a)* fractile and mean hazard curves for each ground motion parameter included in the probabilistic seismic hazard analysis; *(b)* uniform hazard response spectra at hazard exceedance frequencies of interest; *(c)* magnitude-distance deaggregation for the mean hazard; *(d)* seismic source deaggregation; *(e)* ground motion model deaggregation; *(f)* mean magnitude and distance. See Note S-N-7	
SHA-F2	CALCULATE seismic hazard results that are required as input to the seismic PRA quantification (e.g., frequencies of seismic-induced event sequence families modeled in the PRA per the SRs of HLR-SPR-E), including results from the probabilistic seismic hazard analysis, analysis of vertical motions, and analysis of secondary seismic hazards.	
SHA-F3	IDENTIFY, by performing sensitivity studies, the key sources of uncertainty in the probabilistic seismic hazard analysis that may affect the hazard results in a manner that supports Requirements SHA-F4 and SPR-E8. See Note S-N-10	
SHA-F4	PERFORM sensitivity studies to identify the key sources of epistemic uncertainty in the assessment of vertical motions, site response analysis, and the evaluation of secondary hazards performed in SHA-H3 that may affect the quantification results as discussed in Requirement SPR-E8.	

Table 4.3.10.1-8 Supporting Requirements for HLR-SHA-G

For further use in the Seismic Fragility Analysis, the spectral shape shall be based on-site-specific evaluation that considers or incorporates the results of the hazard evaluation. (HLR-SHA-G)

Index No. SHA-G	Capability Category I	Capability Category II
SHA-G1	ENSURE that the horizontal response spectral shape determined in the probabilistic seismic hazard analysis is based on-site-specific evaluations and uses or bounds the characteristic spectral shapes associated with the mean magnitude and distance pairs determined in the probabilistic seismic hazard analysis for the important ground motion levels.	
SHA-G2	JUSTIFY that the methods for determining vertical spectra are appropriate given the current state-of-knowledge. See Note S-N-11	

Table 4.3.10.1-9 Supporting Requirements for HLR-SHA-H

An evaluation shall be performed to assess whether, in addition to the vibratory ground motion, other seismic hazards need to be included in the Seismic PRA. (HLR-SHA-H)

Index No. SHA-H	Capability Category I	Capability Category II
SHA-H1	IDENTIFY fault displacement and secondary seismic hazards associated with vibratory ground motion for the site (e.g., fault displacement, landslides, soil liquefaction, soil settlement, and earthquake-induced external flooding). See Note S-N-12	

Table 4.3.10.1-9 Supporting Requirements for HLR-SHA-H (Cont'd)

An evaluation shall be performed to assess whether, in addition to the vibratory ground motion, other seismic hazards need to be included in the Seismic PRA. (HLR-SHA-H)

Index No. SHA-H	Capability Category I	Capability Category II
SHA-H2	JUSTIFY the basis and methodology used for the screening out of seismic hazards identified by Requirement SHA-H1 (e.g., based on demonstrably conservative assessments) using SCR-2 or SCR-3 in Table 1.10-1. See Note S-N-13	
SHA-H3	For non-flooding related seismic hazards that are not screened out in Requirement SHA-H2, CALCULATE the frequency of levels of hazard parameters used to define the fragility for failure mechanisms of seismic equipment list (SEL) items that may be impacted.	
SHA-H4	For earthquake-induced external flooding hazards that are not screened out in Requirement SHA-H2, SATISFY the applicable SRs of HLR-XFHA-A, HLR-XFHA-B, HLR-XFHA-C, HLR-XFHA-D, HLR-XFHA-E, HLR-XFHA-F, and HLR-XFHA-G, in calculating the frequency of levels of hazard parameters necessary to define the fragility for failure mechanisms of SEL items that may be impacted. See Note S-N-14	

Table 4.3.10.1-10 Supporting Requirements for HLR-SHA-I

The documentation of the Seismic Hazard Analysis shall provide traceability of the work. (HLR-SHA-I)

Index No. SHA-I	Capability Category I	Capability Category II
SHA-I1	DOCUMENT the process used in the Seismic Hazard Analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the data and information that forms the basis and input for the evaluations carried out to develop the Seismic Hazard Analysis inputs, including the seismic source characterization, the ground motion characterization, and the site response; *(b)* the Seismic Hazard Analysis model structure; *(c)* the structured processes used to ensure that the center, body, and range of technically defensible interpretations have been considered; *(d)* the specific methods used for source characterization, ground motion characterization, and local site response analysis; *(e)* the scientific interpretations that are the basis for the Seismic Hazard Analysis inputs and results; *(f)* the sources of uncertainties and related assumptions identified in the Seismic Hazard Analysis that are risk-significant; *(g)* the process to ensure that it meets the requirements herein, if an existing Seismic Hazard Analysis is used; *(h)* the methods for determining vertical spectra; *(i)* the methods for screening and incorporating secondary seismic hazards; and *(j)* the results of the hazard analyses consistent with the SRs of HLR-SHA-F. See Note S-N-15	
SHA-I2	DOCUMENT the sources of model uncertainty (as identified in SHA-F3 and SHA-F4) associated with the Seismic Hazard Analysis. See Note S-N-16	
SHA-I3	For seismic probabilistic risk assessments (SPRA) using a bounding site, DOCUMENT the basis for the selection of the bounding site characteristics that bound the range of sites for which the plant is designed and the justification of the applicability of the bounding site. See Note S-N-9	

4.3.10.2 Objectives and Technical Requirements for Seismic Fragility Analysis (SFR)

The objectives of the Seismic Fragility Analysis are to

(a) develop seismic-fragility information for all the structures, systems, and components (SSCs) in the model;

(b) base the fragility analysis on the seismic response that the SSCs experience at failure;

(c) define the basis and methodology for establishing target fragilities;

(d) incorporate the findings of investigations;

(e) perform seismic-fragility analysis for relevant failure mechanisms; and

(f) document the Seismic Fragility Analysis to provide traceability of the work.

Table 4.3.10.2-1 High Level Requirements for Seismic Fragility Analysis

Designator	Requirement
HLR-SFR-A	The Seismic Fragility Analysis shall address seismic fragilities of SSCs whose failure may affect the frequencies of event sequences and event sequence families modeled in the PRA.
HLR-SFR-B	The Seismic Fragility Analysis shall be based on a seismic response that the SSCs experience at failure.
HLR-SFR-C	The basis and methodologies used to establish the fragility threshold for SSCs shall be defined.
HLR-SFR-D	The Seismic Fragility Analysis shall incorporate the data and findings of investigation(s) of the plant to establish or confirm as-built, as-operated or as-designed, as-intended-to-operate conditions.
HLR-SFR-E	The Seismic Fragility Analysis shall be performed for relevant failure mechanisms affecting the failure modes modeled in the plant response analysis.
HLR-SFR-F	The documentation of the Seismic Fragility Analysis shall provide traceability of the work.

Table 4.3.10.2-2 Supporting Requirements for HLR-SFR-A

The Seismic Fragility Analysis shall address seismic fragilities of SSCs whose failure may affect the frequencies of event sequences and event sequence families modeled in the PRA. (HLR-SFR-A)

Index No. SFR-A	Capability Category I	Capability Category II
SFR-A1	INCLUDE in the scope of the fragility analysis those SSCs and associated failure modes as identified by the plant response analysis. See SRs of HLR-SPR-C See Note S-N-17, S-N-18	
SFR-A2	INCLUDE information relevant to modeling of fragility correlation of SSCs and its basis (e.g., similarity of component construction, location and orientation, and in-structure seismic demand) to support Requirement SPR-B4. See Note S-N-17, S-N-19	

Table 4.3.10.2-3 Supporting Requirements for HLR-SFR-B

The Seismic Fragility Analysis shall be based on a seismic response that the SSCs experience at failure. (HLR-SFR-B)

Index No. SFR-B	Capability Category I	Capability Category II
SFR-B1	ESTIMATE seismic response for use in fragility analysis of the SSCs using the earthquake response spectra shape(s) from SRs of HLR-SHA-G in three orthogonal directions, justifying that any approximations are conservative and appropriate for the plant. See Note S-N-17	CALCULATE realistic seismic response for use in fragility analysis of the SSCs using the earthquake response spectra shape(s) from SRs of HLR-SHA-G in three orthogonal directions, justifying that any approximations are appropriate for the plant. See Note S-N-17, S-N-20
SFR-B2	If scaling of existing response analysis is used, JUSTIFY it based on the adequacy of structural models, foundation characteristics, and similarity of input ground motion. See Note S-N-17, S-N-21	
SFR-B3	USE realistic mathematical structural models to represent the three-dimensional dynamic characteristics of the building structures (e.g., consider stiffness, mass, damping, stress state, directional coupling, rotational inertia, center of mass, discretization, modal frequency response, torsional effects, diaphragm flexibility, structural coupling) for seismic response calculations. See Note S-N-17, S-N-22	
SFR-B4	If median-centered response analysis is performed, ESTIMATE the median response (i.e., structural loads and floor response spectra) and variability in the response. See Note S-N-17, S-N-23	If median-centered response analysis is performed, CALCULATE the median response (i.e., structural loads and floor response spectra) and variability in the response. See Note S-N-17, S-N-23
SFR-B5	For PRAs supporting a specific site, if soil-structure interaction (SSI) effects are considered, ESTIMATE median-centered SSI response and associated uncertainty based on soil properties consistent with site conditions. See Note S-N-24, S-N-25	For PRAs supporting a specific site, if SSI effects are important to structural response, CALCULATE median-centered SSI response and associated uncertainty using site-specific strain-compatible soil properties. See Note S-N-24, S-N-25
SFR-B6	If probabilistic response analysis is performed to calculate structural loads and floor response spectra, ENSURE that the number of simulations done (e.g., Monte Carlo simulation or Latin Hypercube Sampling) is large enough to calculate stable responses. See Note S-N-26	

Table 4.3.10.2-4 Supporting Requirements for HLR-SFR-C

The basis and methodologies used to establish the fragility threshold for SSCs shall be defined. (HLR-SFR-C)

Index No. SFR-C	Capability Category I	Capability Category II
SFR-C1	SPECIFY the basis for defining inherently rugged components. See Note S-N-17, S-N-27	
SFR-C2	SPECIFY the basis and methodologies established for achieving the fragility thresholds defined in Requirement SPR-B5. See Note S-N-17, S-N-28	

Table 4.3.10.2-5 Supporting Requirements for HLR-SFR-D

The Seismic Fragility Analysis shall incorporate the data and findings of investigation(s) of the plant to establish or confirm as-built, as-operated or as-designed, as-intended-to-operate conditions. (HLR-SFR-D)

Index No. SFR-D	Capability Category I	Capability Category II
SFR-D1	CONFIRM that SSCs and associated anchorage that are assigned fragility threshold values satisfy the bases defined in the SRs of HLR-SFR-C. See Note S-N-29, S-N-30	
SFR-D2	EVALUATE the seismic capacity of the as-designed, or as-built, or as-operated, or as-intended-to-operate plant conditions via investigation(s). See Note S-N-31, S-N-32, S-N-33, S-N-34	
SFR-D3	For operating reactors, IDENTIFY seismic vulnerabilities in a manner that ensures the seismic fragility estimations in the SRs of HLR-SFR-E are conservative. See Note S-N-35, S-N-36	For operating reactors, IDENTIFY seismic vulnerabilities in a manner that ensures the seismic fragility calculations in the SRs of HLR-SFR-E are realistic and plant-specific. See Note S-N-35, S-N-36
SFR-D4	For PRAs performed during the pre-operational stage, IDENTIFY seismic vulnerabilities in a manner that ensures the seismic fragility estimations in the SRs of HLR-SFR-E are conservative. See Note S-N-35, S-N-37	For PRAs performed during the pre-operational stage, IDENTIFY seismic vulnerabilities in a manner that ensures the seismic fragility calculations in the SRs of HLR-SFR-E are realistic. See Note S-N-35, S-N-37
SFR-D5	EVALUATE potential functional and structural failure mechanisms, equipment anchorage, and support load path. See Note S-N-17, S-N-38	
SFR-D6	IDENTIFY credible seismic-induced failures (including spray) for the flood sources provided in Requirement SPR-C3.	
SFR-D7	IDENTIFY credible seismic-induced failure for the fire ignition sources provided in Requirement SPR-C4.	
SFR-D8	IDENTIFY credible seismic interactions that may compromise the intended functions of SSCs (see SPR-C6) or operator actions (see SPR-D5). See Note S-N-39	

Table 4.3.10.2-6 Supporting Requirements for HLR-SFR-E

The Seismic Fragility Analysis shall be performed for relevant failure mechanisms affecting the failure modes modeled in the plant response analysis. (HLR-SFR-E)

Index No. SFR-E	Capability Category I	Capability Category II
SFR-E1	For those failure modes identified in Requirement SPR-C6, IDENTIFY conservative bounding failure mechanisms of structures (e.g., sliding, overturning, yielding, and excessive drift) and equipment (e.g., anchorage failure, functional failure, impact with adjacent equipment or structures, and bracing failure). See Requirements SFR-D2, SFR-D3, and SFR-D5 See Note S-N-17, S-N-40	For those failure modes identified in Requirement SPR-C6 that are risk-significant, IDENTIFY relevant and realistic failure mechanisms of structures (e.g., sliding, overturning, yielding, and excessive drift) and equipment (e.g., anchorage failure, functional failure, impact with adjacent equipment or structures, and bracing failure). For non-risk-significant failure modes, use the requirements of Capability Category I (CC-I). See Requirements SFR-D2, SFR-D3, and SFR-D5 See Note S-N-17, S-N-40
SFR-E2	For PRAs conducted on a specific site, for those failure modes identified in Requirement SPR-C6, IDENTIFY conservative bounding failure mechanisms of soil (e.g., liquefaction, slope instability, and excessive differential settlement). See Note S-N-40, S-N-41	For PRAs conducted on a specific site, for those failure modes identified in Requirement SPR-C6 that are risk-significant, IDENTIFY relevant and realistic failure mechanisms of soil (e.g., liquefaction, slope instability, and excessive differential settlement). For non-risk-significant failure modes, use the requirements of CC-I. See Requirements SFR-D2, SFR-D3, and SFR-D5 See Note S-N-9, S-N-40
SFR-E3	ESTIMATE conservative seismic fragilities for the failure mechanisms of interest identified in Requirement SFR-E1 using plant-specific data, or JUSTIFY the use of generic fragility data (e.g., fragility test data, generic seismic qualification test data, and earthquake experience data) or conservative assumptions for the SSCs as being appropriate for the plant. See Note S-N-42, S-N-43	CALCULATE realistic seismic fragilities for the failure mechanisms of interest identified in Requirement SFR-E1 using plant-specific data, or JUSTIFY (e.g., through the calculation of integrated risk metrics defined in Requirement RI-B3) the use of generic fragility data (e.g., fragility test data, generic seismic qualification test data, and earthquake experience data) or conservative assumptions for any SSCs as being appropriate for the plant or by showing no masking or differences in insights. See Note S-N-17, S-N-43
SFR-E4	ESTIMATE contact chatter seismic fragilities for relays and other similar devices that affect SSCs identified in the Systems Analysis. See Requirement SPR-B6 See Note S-N-17, S-N-44, S-N-45	CALCULATE contact chatter seismic fragilities using plant-specific data, or JUSTIFY the use of generic fragility data for relays and other similar devices that affect risk-significant SSCs and are identified in the Systems Analysis. See Requirement SPR-B6 See Note S-N-17, S-N-44, S-N-45
SFR-E5	ESTIMATE seismic fragilities for credible seismic-induced flood sources (see Requirement SFR-D6) and seismic-induced fire ignition sources (see Requirement SFR-D7). See Note S-N-17	CALCULATE seismic fragilities using plant-specific data or JUSTIFY the use of generic fragility data for credible seismic-induced flood sources (see Requirement SFR-D6) and seismic-induced fire ignition sources (see Requirement SFR-D7) that are risk-significant contributors. See Note S-N-17
SFR-E6	IDENTIFY the sources of uncertainty, the related assumptions, and reasonable alternatives of the Seismic Fragility Analysis in a manner that supports Requirement SPR-E8.	
SFR-E7	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Seismic Fragility Analysis. See Note S-N-37, S-N-46, S-N-47	

Table 4.3.10.2-7 Supporting Requirements for HLR-SFR-F

The documentation of the Seismic Fragility Analysis shall provide traceability of the work. (HLR-SFR-F)

Index No. SFR-F	Capability Category I	Capability Category II
SFR-F1	DOCUMENT the process used in the Seismic Fragility Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* seismic response analysis; *(b)* inherently rugged and fragility threshold methodology; *(c)* investigation procedures; *(d)* investigation team composition and member qualification; *(e)* investigation observations and conclusions; *(f)* review of design documents; *(g)* identification of relevant failure mechanisms for each SSC; *(h)* method of capacity evaluation; *(i)* estimation or calculation of fragility parameter values for each SSC modeled (median capacity, logarithmic standard deviation representing the randomness in median capacity, and logarithmic standard deviation representing the uncertainty in median capacity) and sources of information. See Note S-N-48	
SFR-F2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives associated with the Seismic Fragility Analysis identified in SFR-E6. See Note S-N-49	
SFR-F3	For PRAs conducted on bounding sites or during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details or site details associated with the Seismic Fragility Analysis. See SFR-E7 See Note S-N-37	

4.3.10.3 Objectives and Technical Requirements for Seismic Plant Response Model (SPR)

The objectives of the Seismic Plant Response Model are to

(a) develop a plant response model that includes seismically induced initiating events and other failures and the plant's response to them;

(b) develop event sequences based on the plant configuration and the initiating events and failures;

(c) integrate the seismic-hazard analysis and the seismic-fragilities analysis with the systems model to quantify the model;

(d) risk-significant contributors to the seismic-induced event sequences are identified and understood in the context of the plant design, operation, and maintenance;

(e) analysis limitations and uncertainties are understood; and

(f) the Seismic Plant Response Model is documented to provide traceability of the work.

Table 4.3.10.3-1 High Level Requirements for Seismic Plant Response Model

Designator	Requirement
HLR-SPR-A	The Seismic Plant Response Model shall include seismically induced initiating events that cause risk-significant event sequences and/or risk-significant event progression sequences.
HLR-SPR-B	The Seismic Plant Response Model shall include seismic-induced SSC failures, non-seismic-induced SSC failures, unavailabilities, human errors, plant operating states, sources of radioactive material, and multi-reactor effects that can affect the frequencies of seismic-induced event sequence families modeled in the PRA.
HLR-SPR-C	The list of SSCs selected for Seismic Fragility Analysis shall include the SSCs that contribute to in event sequences included in the Seismic Plant Response Model.
HLR-SPR-D	Human actions credited in the Seismic Plant Response Model shall consider seismic-specific challenges to human performance.
HLR-SPR-E	The analysis to quantify the frequency of each seismic-induced event sequence family modeled in the PRA shall integrate the seismic hazard, the seismic fragilities, and the seismic plant response, including uncertainties.
HLR-SPR-F	Documentation of the Seismic Plant Response Model shall provide traceability of the work.

Table 4.3.10.3-2 Supporting Requirements for HLR-SPR-A

The Seismic Plant Response Model shall include seismically induced initiating events that cause risk-significant event sequences and/or risk-significant event progression sequences. (HLR-SPR-A)

Index No. SPR-A	Capability Category I	Capability Category II
SPR-A1	For each plant operating state within the scope of the PRA, use a systematic process to IDENTIFY seismically induced initiating events caused directly by the seismic event. INCLUDE initiating events involving one or more reactors and/or other sources of radioactive material within the scope of the PRA. See Note S-N-50	
SPR-A2	For each plant operating state within the scope of the PRA, use a systematic process to IDENTIFY seismically induced hazard events resulting from secondary hazards (e.g., seismically induced internal flooding, external flooding, and fire), including those identified in HLR-SHA-H that can themselves induce initiating events or fail SSCs modeled in the seismic PRA. INCLUDE initiating events involving damage and release from more than one reactor as well as releases from other sources of radioactive material within the scope of the PRA in this evaluation. See Note S-N-51	
SPR-A3	ENSURE that the initiating events included in the plant response analysis incorporate industry experience (e.g., thorough review of plant-specific response to past seismic events, as well as other available seismic risk evaluations for nuclear plants).	
SPR-A4	INCLUDE in the plant response model the events identified in Requirements SPR-A1, SPR-A2, and SPR-A3 that cause risk-significant event sequence families.	

Table 4.3.10.3-3 Supporting Requirements for HLR-SPR-B

The Seismic Plant Response Model shall include seismic-induced SSC failures, non-seismic-induced SSC failures, unavailabilities, human errors, plant operating states, sources of radioactive material, and multi-reactor effects that can affect the frequencies of seismic-induced event sequence families modeled in the PRA. (HLR-SPR-B)

Index No. SPR-B	Capability Category I	Capability Category II
SPR-B1	USE the event sequences and the systems logic model from the internal event PRA models as the basis of the Seismic Plant Response Model. INCLUDE additional event sequences, as applicable, associated with seismically induced initiating events that are not included in the internal events PRA model such as events that impact two or more reactors or sources of radioactive material. See IE-A16 See Note S-N-52	
SPR-B2	ENSURE that the peer review findings for the internal events and non-seismic hazard PRAs that are relevant to the results of the seismic PRA are resolved and incorporated into the development of the Seismic Plant Response Model. See Note S-N-53	
SPR-B3	INCLUDE seismically induced failures representing the failure modes of interest in the seismic PRA plant response model (e.g., tank rupture, pump failure to start/run, passive system failure). See Requirement SPR-C6	
SPR-B4	MODEL the fragility correlation of seismically induced SSC failures consistently with information provided in Requirement SFR-A2. JUSTIFY the correlation approach used (e.g., by performing sensitivity studies to assess the contribution to the risk results).	
SPR-B5	DEFINE a fragility threshold that, when integrated with the hazard, satisfies SCR-2 in Table 1.10-1. See Note S-N-54	
SPR-B6	Using a systematic process, INCLUDE in the systems analysis the effects of those relays or similar devices whose contact chatter results in the unavailability or spurious actuation of SSCs that are risk-significant contributors to frequencies of event sequence families modeled in the PRA. See Note S-N-55	
SPR-B7	ASSESS the results of the seismic-induced event sequences and SATISFY Requirement SC-A7 at CC-I for success criteria, except where the requirements are not applicable, to confirm that sustained impacts on plant accessibility and emergency response capability do not invalidate the assumed mission time.	ASSESS the results of the seismic-induced event sequences and SATISFY Requirement SC-A7 at Capability Category II (CC-II) for success criteria, except where the requirements are not applicable, to confirm that sustained impacts on plant accessibility and emergency response capability do not invalidate the assumed mission time.
SPR-B8	If new logic is added to the seismic PRA (e.g., new system modeling, new or modified event sequences), SATISFY the requirements of the following at CC-I except where the requirements are not applicable: *(a)* HLR-ES-A and HLR-ES-B for Event Sequence Analysis; *(b)* HLR-SC-A and HLR-SC-B for Success Criteria Development; *(c)* HLR-SY-A and HLR-SY-B for Systems Analysis; *(d)* HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D for Data Analysis; and *(e)* HLR-HR-D for Human Reliability Analysis.	If new logic is added to the seismic PRA (e.g., new system modeling, new or modified event sequences), SATISFY the requirements of the following at CC-II except where the requirements are not applicable: *(a)* HLR-ES-A and HLR-ES-B for Event Sequence Analysis; *(b)* HLR-SC-A and HLR-SC-B for Success Criteria Development; *(c)* HLR-SY-A and HLR-SY-B for Systems Analysis; *(d)* HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D for Data Analysis; and *(e)* HLR-HR-D for Human Reliability Analysis.

Table 4.3.10.3-3 Supporting Requirements for HLR-SPR-B (Cont'd)

The Seismic Plant Response Model shall include seismic-induced SSC failures, non-seismic-induced SSC failures, unavailabilities, human errors, plant operating states, sources of radioactive material, and multi-reactor effects that can affect the frequencies of seismic-induced event sequence families modeled in the PRA. (HLR-SPR-B)

Index No. SPR-B	Capability Category I	Capability Category II
SPR-B9	For any seismic-induced internal flood retained in the seismic PRA, SATISFY the SRs of HLR-FLSN-A, FLESQ-A1, FLESQ-A2, FLESQ-A3, and FLESQ-A4 for Internal Flood Scenarios Development at CC-I except where the requirements are not applicable.	For any seismic-induced internal flood retained in the seismic PRA, SATISFY the SRs of HLR-FLSN-A, FLESQ-A1, FLESQ-A2, FLESQ-A3, and FLESQ-A4 for Internal Flood Scenarios Development at CC-II except where the requirements are not applicable.
SPR-B10	For any seismic-induced internal fire ignition source retained in the seismic PRA, SATISFY SRs of HLR-FPRM-A and HLR-FPRM-B for the Internal Fire Plant Response Model development at CC-I except where the requirements are not applicable.	For any seismic-induced internal fire ignition source retained in the seismic PRA, SATISFY SRs of HLR-FPRM-A and HLR-FPRM-B for the FPRM) development at CC-II except where the requirements are not applicable.
SPR-B11	For any seismic-induced external flooding hazards explicitly retained in the seismic PRA, SATISFY the requirements of the following at CC-I except where they are not applicable: *(a)* External Flood Hazard Analysis SRs of HLR-XFHA-B; *(b)* External Flood Fragility Analysis SRs of HLR-XFFR-A, HLR-XFFR-B, HLR-XFFR-C, and HLR-XFFR-D, to determine the impact of flooding on SSCs; and *(c)* External Flood Plant Response Analysis SRs of HLR-XFPR-A, HLR-XFPR-B, HLR-XFPR-C, HLR-XFPR-D, and HLR-XFPR-E to determine the plant response because of the flood.	For any seismic-induced external flooding hazards explicitly retained in the seismic PRA, SATISFY the requirements of the following at CC-II except where they are not applicable: *(a)* External Flood Hazard Analysis SRs of HLR-XFHA-B; *(b)* External Flood Fragility Analysis SRs of HLR-XFFR-A, HLR-XFFR-B, HLR-XFFR-C, and HLR-XFFR-D, to determine the impact of flooding on SSCs; and *(c)* External Flood Plant Response Analysis SRs of HLR-XFPR-A, HLR-XFPR-B, HLR-XFPR-C, HLR-XFPR-D, and HLR-XFPR-E to determine the plant response because of the flood.
SPR-B12	For all other secondary hazards explicitly retained in the seismic PRA, SATISFY the requirements of the following at CC-I Other Hazards Fragility Analysis SRs of HLR-OFR-A and the Other Hazards Plant Response Analysis SRs of HLR-OPR-B, except where they are not applicable.	For all other secondary hazards explicitly retained in the seismic PRA, SATISFY the requirements of the following at CC-II Other Hazards Fragility Analysis SRs of HLR-OFR-A and the Other Hazards Plant Response Analysis SRs of HLR-OPR-B, except where they are not applicable.
SPR-B13	For multi-reactor sites, ENSURE that the multi-reactor impacts of a seismic event are captured in the plant response model as appropriate. See Note S-N-56	

Table 4.3.10.3-4 Supporting Requirements for HLR-SPR-C

The list of SSCs selected for Seismic Fragility Analysis shall include the SSCs that contribute to in event sequences included in the Seismic Plant Response Model. (HLR-SPR-C)

Index No. SPR-C	Capability Category I	Capability Category II
SPR-C1	USE the internal events systems model developed against Requirement SY-A1 as the basis for developing a SEL to support the fragility analysis. INCLUDE any additional systems that may have been incorporated into the seismic event sequence model in response to SPR-B1. See Note S-N-52, S-N-57	
SPR-C2	INCLUDE in the SEL additional SSCs (e.g., structures, relays, passive components, panels, and cabinets that house PRA components) that may not be explicitly modeled in the internal events model and in other hazard PRAs but require evaluation in the seismic PRA. INCLUDE any additional systems that may have been incorporated into the seismic event sequence model in response to SPR-B1.	
SPR-C3	INCLUDE in the SEL, internal flood sources (as defined in Requirement FLESQ-F2) that have been identified in Requirement SPR-A2.	
SPR-C4	INCLUDE in the SEL, internal fire ignition sources (as defined in Requirement FESQ-F1) that have been identified in Requirement SPR-A2.	
SPR-C5	INCLUDE in the SEL, SSCs that are inducing or are affected by the initiators resulting from the secondary hazards identified in Requirement SPR-A2.	
SPR-C6	For the SSCs identified in Requirements SPR-C1, SPR-C2, SPR-C3, SPR-C4, and SPR-C5, IDENTIFY the failure mode(s) of interest for the fragility analysis to be performed. See Note S-N-58	

Table 4.3.10.3-5 Supporting Requirements for HLR-SPR-D

Human actions credited in the Seismic Plant Response Model shall consider seismic-specific challenges to human performance. (HLR-SPR-D)

Index No. SPR-D	Capability Category I	Capability Category II
SPR-D1	IDENTIFY the human failure events (HFEs) (including recovery actions) from the selected internal events PRA that are relevant in the context of the seismic PRA.	
SPR-D2	For human response actions relevant to the Seismic Plant Response Model, SATISFY the CC-I SRs of HLR-HR-E, except where the requirements are not applicable.	For human response actions relevant to Seismic Plant Response Model, SATISFY the CC-II SRs of HLR-HR-E, except where the requirements are not applicable.
SPR-D3	For definition and specification of HFEs for human response actions identified in Requirement SPR-D2, SATISFY the CC-I SRs of HLR-HR-F, except where the requirements are not applicable.	For definition and specification of HFEs for human response actions identified in Requirement SPR-D2, SATISFY the CC-II SRs of HLR-HR-F, except where the requirements are not applicable.
SPR-D4	For treatment of recovery actions identified in Requirement SPR-D2, SATISFY the SRs of HLR-HR-H, except where the requirements are not applicable.	
SPR-D5	For developing HEPs, SATISFY the CC-I SRs of HLR-HR-G, except where the requirements are not applicable, taking into account relevant seismic related effects on control room and ex-control room post-initiator actions. When addressing influencing factors and the timing considerations covered in Requirements HR-G4, HR-G6, and HR-G8, INCLUDE the effect of the seismic hazard on the control room and ex-control room human actions. See Note S-N-59	For developing HEPs, SATISFY the CC-I SRs of HLR-HR-G, except where the requirements are not applicable, taking into account relevant seismic related effects on control room and ex-control room post-initiator actions. When addressing influencing factors and the timing considerations covered in Requirements HR-G4, HR-G6, and HR-G8, INCLUDE the effect of the seismic hazard on the control room and ex-control room human actions. See Note S-N-59

Table 4.3.10.3-6 Supporting Requirements for HLR-SPR-E

The analysis to quantify the frequency of each seismic-induced event sequence family modeled in the PRA shall integrate the seismic hazard, the seismic fragilities, and the seismic plant response, including uncertainties. (HLR-SPR-E)

Index No. SPR-E	Capability Category I	Capability Category II
SPR-E1	In the quantification of frequencies of seismic-induced event sequence families on a plant-year basis, INTEGRATE the hazard, fragility, and systems analyses in the PRA model.	
SPR-E2	ADDRESS overestimation of risk due to rare event approximations (e.g., where fragilities approach 1.0).	
SPR-E3	ENSURE that the discretization of the hazard curves (e.g., the size and number of bins used to discretize the hazard or other numerical methods used to incorporate the hazard curve in the integration) is appropriate to demonstrate convergence of seismic-induced risk metrics.	
SPR-E4	When quantifying seismic event sequence family frequencies, SATISFY requirements of the following, except where the requirements are not applicable: *(a)* ESQ-A4, ESQ-A6, ESQ-A7; *(b)* ESQ-B1, ESQ-B2, ESQ-B3, ESQ-B5, ESQ-B6, ESQ-B7, ESQ-B8, ESQ-B9, ESQ-B10; *(c)* ESQ-C1, ESQ-C2, ESQ-C3, ESQ-C4, ESQ-C5, ESQ-C6, ESQ-C7, ESQ-C8, ESQ-C9, ESQ-C10, ESQ-C11, ESQ-C12, ESQ-C13, ESQ-C14, ESQ-C15, ESQ-C16, ESQ-C17; *(d)* ESQ-D1, ESQ-D2, ESQ-D3, ESQ-D5, ESQ-D6, and ESQ-D7. See Note S-N-60	
SPR-E5	USE the mean hazard, mean fragilities, and the Systems Analysis to generate point estimates for seismic event sequence family frequencies.	QUANTIFY the mean frequencies of seismic event sequence families and propagate the parameter uncertainty that results from each input (i.e., the seismic hazard, the seismic fragilities, and the Systems Analysis) through the quantification process.
SPR-E6	IDENTIFY in the Seismic Plant Response Model sources of model uncertainty, related assumptions, and reasonable alternatives in a manner that supports Requirement SPR-E8.	
SPR-E7	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Seismic Plant Response Model. See Note S-N-37, S-N-46	
SPR-E8	SATISFY Requirement ESQ-E1 with the additional assumptions identified by each seismic technical subelement in Requirement SHA-F3, fragility analysis, Requirement SFR-E6, and system modeling, Requirement SPR-E6.	

Table 4.3.10.3-7 Supporting Requirements for HLR-SPR-F

Documentation of the seismic response analysis shall provide traceability of the work. (HLR-SPR-F)

Index No. SPR-F	Capability Category I	Capability Category II
SPR-F1	DOCUMENT the process used in the seismic response analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* SEL development and disposition of SSCs; *(b)* the specific modifications made in the internal events PRA model to produce the seismic PRA model and their basis; *(c)* those seismic related influences that affect methods, processes, or assumptions used, as well as the identification and quantification of the HFEs/HEPs; and *(d)* the major outputs of a seismic PRA, such as means and uncertainties of the frequencies of event sequence families, results of sensitivity studies, and risk-significant contributors.	
SPR-F2	DOCUMENT the risk-significant contributors (such as initiating events, event sequences, basic events) to seismic event sequence families in the PRA results summary. DESCRIBE risk-significant event sequence families or risk contributors.	
SPR-F3	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement SPR-E6) associated with the Seismic Plant Response Model.	

Table 4.3.10.3-7 Supporting Requirements for HLR-SPR-F (Cont'd)

Documentation of the seismic response analysis shall provide traceability of the work. (HLR-SPR-F)

Index No. SPR-F	Capability Category I	Capability Category II
SPR-F4	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details or site details associated with the Seismic Plant Response Model. See SPR-E7 See Note S-N-37	
SPR-F5	DOCUMENT limitations in the quantification process that would impact applications.	

4.3.10.4 Peer Review Requirements for Seismic PRA

4.3.10.4.1 Purpose

This Section provides requirements for peer review of the Seismic PRA element of the PRA.

4.3.10.4.2 Peer Review Team Composition and Personnel Qualifications

In addition to the general requirements of Section 6, the peer review team shall have collective knowledge and experience in the areas of systems engineering, seismic hazard, seismic capability engineering, and seismic PRAs, or seismic margin methodologies. The reviewer(s) focusing on the seismic fragility work shall have demonstrated experience in seismic investigation(s) via walkdown(s) of nuclear power plants.

4.3.10.4.3 Review of Seismic PRA Elements to Confirm the Methodology

4.3.10.4.3.1 Seismic Hazard

The peer review team shall evaluate whether the seismic hazard study used in the PRA is appropriately specific to the specific site or bounding site covering a range of sites and has met the relevant requirements of this Standard.

4.3.10.4.3.2 Seismic Fragility Analysis

4.3.10.4.3.2.1 Seismic Response Analysis

The peer review team shall evaluate whether the seismic response analysis used in the development of seismic fragilities meets the relevant requirements of this Standard. Specifically, the review should focus on the input ground motion (i.e., spectrum or time history), structural modeling, including soil-structure interaction effects, parameters of structural response (e.g., structural damping and soil damping), and the reasonableness of the calculated seismic response.

4.3.10.4.3.2.2 Seismic Walkdown

The peer review team shall review the seismic walkdown of the plant to ensure the reasonableness of the findings of the seismic review team on screening, seismic spatial interactions, and the identification of critical failure mechanisms.

For PRAs on plants prior to operation, if plant walkdown(s) is not possible, the peer review team should review the findings of the following information via interviews and reviews (e.g., tabletop reviews, computerized simulations) with engineering personnel to ensure the reasonableness of the findings of the seismic review team on screening, seismic spatial interactions, and the identification of critical failure mechanisms.

4.3.10.4.3.2.3 SSC Fragility Analysis

The peer review team shall evaluate whether the methods and data used in the fragility analysis of SSCs are adequate for the purpose.

4.3.10.4.3.3 Seismic Plant-Response Analysis

4.3.10.4.3.3.1 Seismic-Induced Initiating Events

The peer review team shall evaluate whether the seismically induced initiating events are properly identified and analyzed.

4.3.10.4.3.3.2 Seismic Event Sequence Analysis

The peer review team shall evaluate whether, in the systems analysis, the SSCs are properly modeled and the event sequences are properly analyzed and quantified. The review team shall ensure that the seismic equipment list is reasonable for the plant or design considering the reactor type, design vintage, and specific design.

4.3.10.4.3.3.3 Seismic Quantification

The peer review team shall evaluate whether the seismic quantification method used in the seismic PRA is appropriate and provides all the results and insights needed for risk-informed decisions. The review shall focus on the event sequence, on the event sequence family frequency estimates and uncertainty bounds, and on the risk-significant contributors.

4.3.10.5 References for Seismic PRA

The following is a list of publications referenced in this Standard.

[S-1] U. S. Nuclear Regulatory Commission (E Leeds and M Johnson) Letter to All Power Reactor Licensees et al., "Request for Information Pursuant to Title 10 of the Code of Federal Regulations 50.54(f) Regarding Recommendations 2.1, 2.3 and 9.3 of the Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident," March 12, 2012

[S-2] EPRI (2004, 2006) "Ground-Motion Model (GMM) Review Project," Electric Power Research Institute, EPRI Report 3002000717, 2013

[S-3] American Society of Civil Engineers, "Seismic Analysis of Safety-Related Nuclear Structures and Commentary," ASCE 4-16, April 2017

[S-4] "Seismic Fragility and Seismic Margin Guidance for Seismic Probabilistic Risk Assessments," EPRI 3002012994, Electric Power Research Institute, Palo Alto, California, USA, 2018

[S-5] R. J. Budnitz, G. S. Hardy, D. L. Moore, and M. K. Ravindra, NUREG/CR-7237, "Correlation of Seismic Performance in Similar SSCs (Structures, Systems, and Components)," December 2017

[S-6] Regulatory Guide 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," Nuclear Regulatory Commission, March 2007

[S-7] NTS Engineering, RPK Structural Mechanics Consulting, Pickard, Lowe & Garrick, Woodward Clyde and Consultants, and Duke Power Company, "A Methodology for Assessment of Nuclear Power Plant Seismic Margin," Electric Power Research Institute, Palo Alto, CA. EPRI NP 6041-SL, Revision 1, 1991

[S-8] Electric Power Research Institute, "Seismic Evaluation Guidance-Screening, Prioritization and Implementation Details (SPID) for the Resolution of Fukushima Near-Term Task Force Recommendation 2.1: Seismic," EPRI Report 1025287, Revision 0, 2013

[S-9] "High Frequency Program: Application Guidance for Functional Confirmation and Fragility," Report 3002004396, Electric Power Research Institute, 2015

[S-10] "A Preliminary Approach to Human Reliability Analysis for External Events with a Focus on Seismic," Report 1025294, Electric Power Research Institute, 2012

[S-11] Electric Power Research Institute, "Seismic Probabilistic Risk Assessment Implementation Guide," EPRI Report 3002000709, 2013

[S-12] J. P. Ake, C. Munson, J. Stamatakos, M. Juckett, K. Coppersmith, and J. Bommer, "Updated Implementation Guidelines for SSHAC Hazard Studies," Report NUREG-2213, U.S. Nuclear Regulatory Commissions, 2018

[S-13] A. M. Kammerer and J. P. Ake, "Practical Implementation Guidelines for SSHAC Level 3 and 4 Hazard Studies," Report NUREG-2117, U.S. Nuclear Regulatory Commission, 2012

[S-14] R. J. Budnitz, D. M. Boore, G. Apostolakis, L. S. Cluff, K. J. Coppersmith, C. A. Cornell, and P. A. Morris, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts," Report NUREG/CR-6372, U.S. Nuclear Regulatory Commission, 1997

[S-15] A. M. Kammerer, R. J. Budnitz, N. C. Chokshi, and K. Coppersmith, "Proposed Risk-Informed Seismic Hazard Periodic Reevaluation Methodology for Complying with DOE Order 420.1C," INL/EXT-15-36510," Revision 1, Idaho National Laboratory Report INL/ EXT-15-36510 Revision 1, Idaho Falls, November 2015

[S-16] NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," Nuclear Regulatory Commission, June 1991

[S-17] R. K. McGuire, W. J. Silva, and C. J. Costantino, NUREG/CR-6728, "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines," Nuclear Regulatory Commission, October 2001

[S-18] NUREG 2115, CEUS-SSC, "Technical Report: Central and Eastern United States Seismic Source Characterization for Nuclear Facilities," EPRI, Palo Alto, CA, U.S. DOE, and U.S. NRC, 2012

[S-19] Regulatory Guide 1.132, "Site Investigations for Foundations of Nuclear Power Plants," Nuclear Regulatory Commission, October 2003

[S-20] Regulatory Guide 1.138, "Laboratory Investigations of Soils and Rocks for Engineering Analysis and Design of Nuclear Power Plants," Nuclear Regulatory Commission, December 2014

[S-21] "Surry Seismic Probabilistic Risk Assessment Pilot Plant Review," Report 1020756, Electric Power Research Institute, 2010

[S-22] Seismic Qualification Utility Group (2001), "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment," Revision 3A, Copyright © 2001 SQUG

[S-23] NUREG/CR-6728, R. K. MCGUIRE et al., "Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-Consistent Ground Motion Spectra Guidelines," U.S. Nuclear Regulatory Commission, 2001

NONMANDATORY APPENDIX S: NOTES AND EXPLANATORY MATERIAL FOR SEISMIC PRA

S.1 NOTES ASSOCIATED WITH SEISMIC PRA

Table S-1 Notes Supporting Seismic PRA Requirements

Number	Notes
S-N-1	The description of a bounding site can be the identification of an existing site if that site bounds the hazard for other sites under consideration. See SHA-A1
S-N-2	The Senior Seismic Hazard Analysis Committee (SSHAC) report [S-14] and related guidance [S-13] provide the defined process for conducting a probabilistic seismic hazard analysis that produces a model that represents the center, body, and range of the technically defensible interpretations, as defined in those references. These references have identified and provided guidance for four levels of hazard analysis. The SSHAC study level should be chosen consistent with the intended application following the guidance in [S-13]. See SHA-A2
S-N-3	The lower bound magnitude specified should be consistent with current practice. RG 1.208 [S-6] provides one acceptable approach to establishing a lower bound magnitude for use in the hazard analysis. See SHA-A6
S-N-4	The example approaches method for establishing the number of standard deviations (ϵ) to be included in the analysis of ground motion prediction equations can be found in RG 1.208 [S-6]. See SHA-A7
S-N-5	Guidelines as to when an existing study should be refined or replaced are provided in [S-13]. See SHA-B1
S-N-6	This requirement is particularly relevant with regional seismic source characterization and ground motion characterization (GMC) models that are used as part of a probabilistic seismic hazard analysis. Revisions to key inputs to a Seismic Hazard Analysis and the analysis results occur periodically. Thus, one of the important features of conducting a probabilistic seismic hazard analysis is to ensure that these key inputs represent the currently available data, models, and methods and that the probabilistic seismic hazard analysis represents the center, body, and range of technically defensible interpretations. Guidelines as to when an existing study should be refined or replaced are provided in [S-13]. See SHA-B4
S-N-7	References [S-13] and [S-14] provide a structured approach for conducting the probabilistic seismic hazard analysis consistent with the level of analysis defined in HLR-SHA-A. These references also provide a process for producing a seismic sources model that represents the center, body, and range of the technically defensible interpretations. See SHA-C2, SHA-D2, SHA-F1
S-N-8	The identification and inclusion of uncertainty is required because seismic sources are numerically characterized based on alternative interpretations and conceptual models that can include alternative geometries, alternative estimates of maximum earthquake magnitude, and alternative earthquake recurrence models and parameters [S-13] and [S-14]. See SHA-C3
S-N-9	This SR is not applicable to PRAs performed for a specific site. See SHA-E2, SHA-E4, SHA-E6, SHA-I3, SFR-E2

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-10	For site-specific studies that do not use regional models, sensitivity studies and intermediate results provide important information to reviewers about how some of the key assumptions affect the final results of a complex seismic hazard process. Examples of useful sensitivity studies include an evaluation of alternate schemes used to assign weights to the individual expert models and an evaluation of the way different experts make different assignments of the regional seismicity to different zonation maps. Where regional studies are used, it is important to develop documentation describing how the sensitivity and uncertainty information from in the regional model's documentation provide insights for the site being analyzed. See SHA-F3
S-N-11	RG 1.208 [S-6] provides one approach to establishing vertical-to-horizontal (V/H) spectral ratios, which can be combined with the appropriate horizontal spectra to derive vertical spectra. EPRI [S-4] provides guidance for developing mean V/H ratios for a range of site conditions (rock and soil) and levels of ground motion. See SHA-G2
S-N-12	SHA-H1 addresses development of the list of secondary hazards requiring further evaluation for the seismic PRA. The list is developed by first starting with a very broad list of possible secondary hazards and then screening in or out on a hazard-by-hazard basis. The appropriate approach used to justify the basis and methodology used for screening out other direct seismic hazards (e.g., fault displacement) or secondary hazards caused by vibratory ground motions (e.g., soil liquefaction, soil settlement, and earthquake-induced external flooding) is hazard- and site-specific. Justification may be based on publicly available literature and prior hazard studies. Qualitative screening criteria can be applied for cases where the hazard is physically not possible or is exceedingly rare (e.g., triggering of secondary hazard $<10^{-7}$/yr) as assessed by a demonstrably conservative deterministic analysis/assessment. See SHA-H1
S-N-13	If site conditions make it necessary to include other seismic hazards, the objective of the subsequent analysis is to estimate the frequency of hazard occurrence as a function of amplitude or intensity of the parameter appropriate for the failure mechanism(s) of interest. Because understanding the risk implications of other seismic hazards requires additional analysis within a PRA framework, the approach used to analyze additional hazards, as well as the parameters assessed, should be integrated with the fragility and model analysis activities. The figure in Section S.2.1 shows how SHA-H1 and SHA-H2 flow from and to other SRs. There are two points where screening of the hazard from the seismic PRA model is considered once it has been screened into further evaluation (under SHA-H2) by SHA-H1. This approach starts with the SEL and screens the SSCs out on an SSC-by-SSC basis. Screening type 1 removes SSCs that cannot be impacted due to location (e.g., the SSC is a location that is not susceptible to liquefaction). Screening type 2 removes SSCs that can be demonstrated to have sufficient capacities within the context of the seismic PRA given the hazard levels (e.g., the convolution of the fragility and hazard curves demonstrates a sufficiently low probability of failure). Screening types 1 and 2 require interactions between Seismic Hazard Analysis, Seismic Fragility Analysis and Seismic Plant Response Model analysts and should be considered within the framework in Table 1.10-1. Requirements for seismically induced internal fire and flooding are addressed in SFR-D3, SFR-D5, and SPR-B9. See SHA-H2

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-14	This SR is focused on earthquake-induced external flooding hazards (e.g., upstream dam failure) that are not screened out in SHA-H2 and refers the reader to the applicable SRs of HLR-XFHA-A, HLR-XFHA-B, HLR-XFHA-C, HLR-XFHA-D, HLR-XFHA-E, HLR-XFHA-F, and HLR-XFHA-G, in calculating the frequency of levels of hazard parameters necessary to define the fragility for failure mechanisms of SEL items that may be impacted. This SR is required because there is an interface between seismic hazard and flooding hazard inherent in the earthquake-induced external flooding hazards and there is, therefore, a need for consistency between the Sections and clarity as to where the SRs ultimately sit. See SHA-H4
S-N-15	The project documentation is the fundamental basis for reviews and users to understand. Guidance as found in References [S-13] and [S-14] should be consulted regarding expectations for this documentation: *(a)* the process used to develop probabilistic seismic hazard analysis inputs and perform probabilistic seismic hazard analysis computations; *(b)* the data that were available and used in the evaluation process; *(c)* how the data, models, and methods of the larger technical community were integrated and considered in developing the probabilistic seismic hazard analysis inputs; *(d)* the elements that make up the probabilistic seismic hazard analysis input model and their technical bases; *(e)* how uncertainties were modeled and quantified, and how these capture the center, body, and range of technically defensible interpretations; and *(f)* the probabilistic seismic hazard analysis results and instructions for their use. See SHA-I1
S-N-16	The level of effort for developing probabilistic seismic hazard analysis documentation depends on whether and to what extent existing probabilistic seismic hazard analysis information is being used for the probabilistic seismic hazard analysis. For example, sites that use the seismic source model from NRC/DOE/EPRI [S-18] or the ground motion model from EPRI [S-2] can take advantage of the significant documentation available for those projects. For those sites where a new probabilistic seismic hazard analysis is performed, particularly for a SSHAC Level 3 probabilistic seismic hazard analysis, a significant effort may be necessary to develop adequate probabilistic seismic hazard analysis documentation. Guidance as found in References [S-13] and [S-14] should be consulted regarding expectations for this documentation. It should be recognized by all parties involved in the probabilistic seismic hazard analysis study (sponsor, analyst, peer reviewer, fragility analyst, risk analyst, regulator, and members of the public) that expectations for developing adequate probabilistic seismic hazard analysis documentation can be a difficult and controversial issue. Probabilistic seismic hazard analysis documentation is intended to make the probabilistic seismic hazard analysis tractable from process to inputs to results to sensitivities. In addition to the guidance above, to meet the expectations of SHA-I1, this documentation should do the following: *(a)* describe the roles and responsibilities of all project participants; *(b)* provide sufficient information to understand which parts of the probabilistic seismic hazard analysis inputs (e.g., dominant seismic source, ground motion attenuation model) dominate the seismic hazard at the annual frequencies of exceedance important to the project; *(c)* provide sufficient information showing the sensitivity of hazard results to the uncertainty in key parameters and variation in the hazard due to the changes in parameter values considered in the hazard assessment; *(d)* provide sufficient tabulated data and data files that facilitate the ability to understand hazard inputs and to examine specific parameter assessments or scientific interpretations; *(e)* document any peer review of the probabilistic seismic hazard analysis, including a summary of the whether the peer review was participatory, and the comments and conclusions of the peer reviewers or panel.

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-16 (Cont'd)	While the probabilistic seismic hazard analysis documentation needs to meet the general expectations described above, the following specific probabilistic seismic hazard analysis results should be tabulated and provided with the final probabilistic seismic hazard analysis documentation to meet the state-of-practice for SHA-I1: *(a)* A tabulated set of inputs for both seismic source models and ground motion models used with the probabilistic seismic hazard analysis. The probabilistic seismic hazard analyst is encouraged to review Reference [S-13] and the expectations for developing a probabilistic seismic hazard analysis input document. The tabulated set of inputs should be supplemented with sufficient graphical information to display the seismic source and ground motion models inputs. If existing probabilistic seismic hazard analysis inputs are used and the documentation of these inputs is readily available in published reports, this information does not have to be repeated in the probabilistic seismic hazard analysis report for the site of interest. *(b)* Seismic hazard curves for horizontal ground motion for peak ground acceleration and spectral frequencies. A sufficient number of spectral frequencies should be used to enable robust determination and tabulation of Uniform Hazard Response Spectra (UHS). Mean and fractile (e.g., 5th, 15th, 50th, 85th, and 95th) hazard curves should be provided to clearly display the quantification of uncertainties. The seismic hazard curves should represent the reference site condition associated with the ground motion model used for the probabilistic seismic hazard analysis. *(c)* UHS at representative mean annual frequencies of exceedance such as 10^{-3}, 10^{-4}, and 10^{-5}. If the UHS from the probabilistic seismic hazard analysis is insufficient to fully describe the spectral shape, the approach used to develop smoothed UHS should be documented along with a tabulated set of smoothed UHS for the representative mean annual frequencies of exceedance. *(d)* Deaggregation of the hazard for an appropriate suite of distance and magnitude bins (see Reference [S-6]). *(e)* Vertical UHS including a tabulated set of V/H ground motion ratios if these are used to derive vertical ground motions for horizontal ground motions. Deaggregation in terms of epsilon will also be provided. *(f)* As appropriate, input model used for site response analysis included tabulated values of shear wave velocity, thickness, and density for all layers. If multiple profiles are modeled, this information should be provided for all profiles. Additionally, tabulated values for all strain dependent properties should be provided for all layers, including shear modulus and damping degradation with shear strain. *(g)* As appropriate, input ground motions used to perform the site response analysis, including tabulated values of these motions. *(h)* As appropriate, site amplification factors at each spectral frequency (and peak ground acceleration) modeled in the site response analysis. For each input motion, a tabulated set of mean and (log) standard deviations should be provided. *(i)* Seismic hazard curves and UHS for horizontal ground motion for peak ground acceleration and spectral frequencies at the reference control point from the site response analysis. The documentation should clearly describe the approach to deriving the seismic hazard curves. *(j)* Horizontal and Vertical Ground Motion Response Spectra at the reference control point. A tabulated set of values should be provided. *(k)* As appropriate, for cases with multiple control points, Foundation Input Response Spectra at each control point. A tabulated set of values should be provided. *(l)* A tabulated set of any seismic hazard curves if these are used for seismic risk quantification purposes. Either in graphical or tabulated form, results displaying the most significant contributors to the seismic hazard at the site. This assessment could include an assessment of the variance contribution for each of the major probabilistic seismic hazard analysis inputs to the total variance modeled in the probabilistic seismic hazard analysis. See SHA-I2

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-17	Although a site-specific and plant-specific analysis of the fragilities of SSCs provides the most realistic estimates, for the pre-operational stage PRA, it may be acceptable to develop fragility estimates on some other basis (e.g., generic information), provided that the uncertainties introduced are acceptable for the application. For PRAs performed during the pre-operational stage, decoupling the fragility analysis from the hazard analysis to the extent practical allows for the fragility analysis to bounding a range of sites. It may be necessary to use generic data for this purpose. See SFR-A1, SFR-A2, SFR-B1, SFR-B2, SFR-B3, SFR-B4, SFR-C1, SFR-C2, SFR-D5, SFR-E1, SFR-E3, SFR-E4, SFR-E5
S-N-18	In a seismic PRA, it is customary for the systems analyst to define the initial list of SSCs for fragility analysis. This list generally includes failure modes of interest (e.g., loss of operability or failure of pressure boundary), building location of component, and component description. Therefore, it is expected for the fragility analyst to perform a fragility assessment, whether by calculation or judgment, for the SSCs and relevant failure modes defined by the systems analyst in the SEL. The importance of SFR-A1 stems from the need to ensure that there is consistency between what the systems analyst defines as the scope for the fragility analyst such as, for example, providing fragility curves for the failure mode modeled in the seismic PRA. In the context of fragility analysis, the term failure mechanism refers to the seismic-induced failure of a component that leads to the failure mode defined by the systems analyst. Failure of valves provides a good example of this mechanism-mode relationship: first, the systems analyst defines the failure mode as the valve failing to open on demand, whereas the seismic fragility analyst then defines the failure mechanism induced by an earthquake to result in such a failure mode as either excessive binding of the valve yoke or malfunction of the operator. Experience has shown that the number of SSCs for which fragilities are required will most likely vary throughout the duration of the seismic PRA project. For example, new SSCs may be added to the scope of work resulting from investigation observations. On the other hand, there could be instances where the initial SEL is reduced because SSCs are no longer credited in the seismic PRA model. It is recommended to document this process on how the scope of fragility calculations evolves throughout the seismic PRA project. See SFR-A1
S-N-19	This SR ensures that relationships between failure probabilities of individual SSCs are appropriately modeled in the seismic PRA. Two SSCs are independent if the probability of both failing together is the product of their individual failure probabilities. When failures of two SSCs are not independent, then the two SSCs are said to be dependent. The relationship between the two SSCs underlying this dependency could be causal (i.e., failure of one SSC triggers the failure of the other) or otherwise, and this relationship should be represented in the plant response model. An example applicable to seismic PRA is a component mounted in a structure, wherein collapse of the structure causes failure of the component. Causal dependencies are usually directly represented in the plant response model through appropriate logic gates. Non-causal dependencies may be represented in the plant response model through some combination of systems logic, grouping, and fragility correlation. Fragility correlation is a dependency between two SSCs' ground motion capacities that can be represented by a linear relationship. Perfect correlation between two SSC fragilities occurs when the SSC ground motion capacities[7] are linearly proportional. As such, for perfectly correlated SSCs, the conditional probability of failure of one SSC given the failure of the other is higher than its original (unconditional) failure probability. SSCs with uncorrelated fragilities have no linear dependence between their ground motion capacities, and knowledge about failure of one SSC does not inform the failure probability of the other. Many situations occur somewhere between these two extremes,

[7] A random variable, whose cumulative distribution function is usually defined by a double-lognormal curve and the fragility parameters: median ground motion capacity, logarithmic standard deviation for randomness, and logarithmic standard deviation for uncertainty.

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-19 (Cont'd)	wherein a partial correlation is said to exist between the two fragilities. Current seismic PRA practice idealizes partial correlations as either perfectly correlated or uncorrelated, whichever of the two is more appropriate. A rigorous treatment of partial correlations is presented in NUREG/CR-7237 [S-5]. The determination of whether two or more SSCs' fragilities are correlated, and the degree of correlation, requires a comparison of the following: *(a)* seismic demands associated with the SSC failure mechanisms (e.g., acceleration demands due to seismic shaking, displacement demands in case of seismic spatial interaction); *(b)* seismic capacity associated with the SSC failure mechanism (e.g., relay chatter acceleration capacity, anchor capacity). SSCs of similar construction (e.g., equipment type, materials, physical configuration), installed in a similar fashion (e.g., directional orientation, anchorage type), and located near each other (e.g., in the same area, floor, and building) are expected to have similar failure mechanisms, seismic demands, failure modes, and seismic capacities. As such, the seismic fragilities for these SSCs are typically modeled as perfectly correlated. Consequently, information pertinent to similarity between component construction, installation, and location should be reviewed and included in the basis for decisions regarding fragility correlation in the systems risk model. However, determining the appropriate idealization (uncorrelated or perfectly correlated) may not be straightforward in many cases. Consider two dissimilar SSCs with similar dominant frequencies located next to each other. If the seismic fragility variabilities of the two SSCs are almost completely dominated by variability in the seismic demands (say, due to large variability in the soil properties underlying the building), and the difference in the seismic capacities associated with the failure mechanisms (which may or may not be similar) is small, then significant partial correlation may exist between the two SSC fragilities. Communication between the fragility and systems analysts is important in such situations to ascertain the appropriate modeling idealization: e.g., if modeling the two SSCs as uncorrelated produces conservative risk results with negligible impact on the overall risk insights (as determined from sensitivity analyses), it may be appropriate to ignore the non-trivial partial correlation between the two SSCs. See SFR-A2
S-N-20	The relationship between the structure response and the amplitude of seismic input motion is inherently nonlinear across the range of accelerations considered in a seismic PRA. Seismic PRAs are often quantified for spectral accelerations or peak ground accelerations (PGAs) ranging from 0 g to 5 g or greater. Due to the nonlinear nature of soil and structure behavior, the response at any given acceleration in this range will be a function of both the input ground motion and its effect on the nonlinear characteristics of the system. Ideally, structure response analyses would be performed at several different input levels to determine the varying seismic demands on SSCs across the full range of ground motion levels for which the probability of failure of any credited SSC contributes to the overall risk. In practice, however, some simplification is generally warranted. Therefore, an elastic structure response analysis is typically performed to develop in-structure response spectra (ISRS) for one input level, defined by the "reference earthquake" (RE), and the ISRS are linearly scaled to other input levels as a simplifying approximation. The intent of SFR-B1 is to ensure that the seismic demands used in fragility analysis in HLR-SFR-E, including any corresponding simplifications, are sufficiently realistic (or conservative for CC-I) not to significantly bias the overall seismic PRA results and risk insights. Many different response analysis approaches could potentially meet this intent, and this nonmandatory appendix does not endorse any one approach as preferred over any other. Rather, the commentary below outlines several elements that should be considered and briefly describes an approach that has been used in past seismic PRAs to meet the intent of SFR-B1. The following elements should be considered in the seismic response analysis to ensure adequate realism:

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes

S-N-20 (Cont'd)	*(a)* Nonlinearity in seismic response with increasing ground motion input levels (e.g., closing of gaps, building-to-building interaction, strain-compatible soil properties, degradation effects such as concrete cracking, steel yielding, increased damping, reduced stiffness). Potential nonlinear effects should be identified, and nonlinearities that are found to have a significant effect on the seismic PRA results should be directly addressed in the seismic PRA.
	(b) RE/hazard range of interest (HROI). The specific (or range of) ground motion level(s) for which seismic PRA results and risk insights are most sensitive to seismic demands on SSCs should be identified and understood to inform the selection of analysis simplifications for seismic response analysis. The range(s) of ground motion levels should be selected and used such that it does not introduce significant bias in the seismic PRA results.
	(c) The response spectrum shape used to define seismic input to the response analysis. In general, the Uniform Hazard Response Spectrum (UHRS) shape will differ somewhat at different ground motion input levels. The shape of the input response spectra should be defined to ensure it does not introduce significant bias in the seismic PRA results.
	(d) The ground motion reference parameter [e.g., PGA or spectral acceleration (SA)] and control point. Most seismic PRAs express fragilities in terms of PGA at a specific control point (e.g., at the ground surface, top of rock, basemat of the reactor building), and then the risk quantification convolves the fragilities with the PGA hazard defined at the same control point. It can also be acceptable, and sometimes preferred, to express fragilities in other terms, such as the average ground spectral acceleration over an important frequency range. Whatever parameter and control point is selected, they must be used consistently throughout the seismic PRA within the Seismic Hazard Analysis, Seismic Fragility Analysis, and Seismic Plant Response Model technical elements.
	(e) Input time histories. Seismic PRAs typically use time history analysis to develop ISRS for input to equipment seismic fragilities. The input ground motion time histories must be selected, developed, and/or conditioned in such a way to preclude introducing significant bias into the seismic PRA results.
	Further discussion is provided below on these elements as they pertain to meeting the intent of SFR-B1.
	Structure response analyses in seismic PRAs to date have typically been performed at a single input level and then linearly scaled to estimate responses at other levels. To minimize bias introduced by this linear approximation, the analyses are performed using soil/structure properties (stiffness and damping) and nonlinear behavior (e.g., boundary conditions, building-to-building interaction) corresponding to a RE ground motion input (spectrum shape and level). The RE should be selected carefully and subsequently validated when following this simplified approach to structural response analyses. One reasonable approach for selecting a RE is as follows:
	(a) Estimate one or both of the following based on available information prior to performing seismic response analyses:
	    (1) Estimate the event sequence frequencies based on the best available information prior to performing the seismic PRA. The estimate should consider the seismic design criteria, prior seismic PRAs (site-specific or from similar plants), and the latest site-specific seismic hazard estimate relative to prior hazard estimates.
	    (2) Estimate a plant-level fragility for radiological release based on the best available information prior to performing the seismic PRA. The plant level fragility is the conditional probability of the damage state as a function of input level (e.g., the seismic radiological release frequency plant-level fragility is the conditional radiological release probability as a function of input level). The fragility estimate should consider the seismic design criteria, past evaluations (seismic margin, seismic evaluation process, individual plant examination of external events, expedited seismic evaluation process, etc.), and the latest hazard. Based on experience, logarithmic standard deviations (β) for a plant level fragility are typically in the range of 0.3 to 0.5.

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-20 (Cont'd)	*(b)* Use convolution to relate the plant level fragility (or candidate fragilities), the site-specific seismic hazard, and the seismic risk. The convolution is typically performed via numerical integration. Trial and error iterative approaches can be used to test candidate seismic fragilities and/or seismic risks (depending on choice of 1a and 1b and information available) to arrive at an RE definition considering all site-specific insights. The resulting risk and/or fragility will necessarily depend on engineering judgment and should be validated and potentially adjusted as the seismic PRA progresses. *(c)* Inspect the convolution results across the range of input levels considered and use this information to identify the input level(s) that contributes most significantly to seismic risk. The significance of risk contribution from various input levels can be judged several ways, such as the following examples: *(1)* The level at which the cumulative risk reaches 50% of the total. *(2)* The level where the integrand of the convolution integral is maximized. The integrand can be considered a "risk density" (units are $(\text{yr-g})^{-1}$). *(3)* The level where the slope of the plant level fragility curve is greatest. *(d)* Select a UHRS with an annual exceedance probability that is reasonably aligned with the dominant input level. The UHRS selected is the RE. Conventionally, the RE is selected as either the 10^{-4} or 10^{-5} UHRS. If the dominant input level lies between the 10^{-4} and 10^{-5} UHRS, then the RE is often instead defined as the ground motion response spectra per ASCE/SEI [S-3]. For very low-hazard and/or seismically robust plants, the dominant input level could be closer to the 10^{-6} UHRS. *(e)* As initial Seismic PRA results become available, the risk-dominant input level should be evaluated to assess whether it is reasonably aligned with the RE. If it is anticipated that the final seismic PRA results will show significant misalignment between the dominant input level and the RE, then the effect of the misalignment should be evaluated. Examples where the seismic PRA results may motivate potential adjustment to the RE selection and/or potential extension or enhancement of the structure response analyses include the following: *(1)* Existence of low-capacity/high-importance SSCs. In these cases, the RE may be at a higher ground motion than the failure level of such SSCs, such that the estimated fragility of these SSCs could be unrealistic. *(2)* Existence of high-capacity/high-importance SSCs. In these cases, the RE may be at a lower ground motion than the failure level of such SSCs, such that the estimated fragility for these SSCs could be unrealistic. *(3)* Broad range of risk contribution. In these cases, the seismic risk may be governed by a wide range of relatively equally significant SSC fragilities, such that the selection of a single RE may not be representative of the risk contribution for each. *(4)* Low hazard and/or robust plant. In these cases, the SSC fragilities can be sufficiently high such that meaningful probability of failure coincides only with extremely rare earthquakes, and risk contribution can end up "smeared" across a wide range of large ground motion levels. *(5)* Nonlinear "cliff-edge" effects. In these cases, use of a RE defined at a ground motion level lower than the initiation of significant nonlinear effects resulting in SSC fragilities greater than this level could be unrealistic. Examples include building-to-building impact, nuclear steam supply systems support conditions in a PWR, onset of foundation sliding, and others. Once the RE or HROI selection is made, then other technical decisions related to the topics introduced earlier in this text may follow, as discussed in the paragraphs below. If an alternative simplification for seismic response analysis is taken (i.e., rather than a single RE/HROI), then similar (but perhaps broader) decisions should still be made, with the following discussion still relevant. The RE selection defines the degradation levels (e.g., strain-compatible soil properties, building modeling parameters such as stiffness and damping, contact when gaps close) used in seismic response analyses. Structure and soil properties tend to degrade when subjected to higher levels of ground motion. In some beyond-design-basis events, the

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-20 (Cont'd)	behavior of the soil-structure system will be dominated by non-linear soil effects and/or significant concrete cracking. Therefore, modeling inputs used in the seismic response should correspond to the level of structure and soil material degradation expected for the RE level and spectral shape. SFR-B1 requires that the RE shape, input level, and corresponding degradation states of soil and building models do not introduce significant error or bias in the seismic PRA results. Another important aspect of SFR-B1 is the selection of the ground motion reference parameter (e.g., PGA or SA) and control point used in the convolution of fragilities and the plant seismic hazard curve. The two common ground motion parameters used in seismic PRAs are the PGA and average SA. Average SA is considered a good indicator of earthquake damaging effects and is sometimes preferred as the ground motion parameter for fragility analysis. However, PGA has historically been used in more seismic PRAs, and therefore may be more familiar to the hazard, fragility, and systems engineers. It is essential, however, that whichever parameter is chosen, it be used consistently throughout the seismic PRA process. Similarly, the seismic hazard curves, RE ground motion, and fragilities must be defined at a common control point that is used consistently in the response analysis, fragility analysis, and risk quantification. It would be erroneous, for example, to express fragilities in terms of the PGA at the ground surface, and then to convolve them with PGA hazard curves defined at a control point at depth within the underlying soil/rock. When input to seismic response analyses are defined by time histories, the time histories should be developed to be consistent with the selected ground motion input level, control point, and spectral shape. Several industry guidance documents provide guidance for creating artificial time histories and/or selecting and conditioning natural seeds. For example, ASCE/SEI 4-16 [S-3] provides design criteria for time history matching. The guidance and commentary in the industry literature should be considered when developing time histories to ensure and justify that the time histories used in the response analysis do not introduce significant error or bias into the seismic PRA results. SFR-B1 uses different action verbs for Capability Categories I and II: ESTIMATE for CC-I and CALCULATE for CC-II. Here, it is important to distinguish between "estimated response" and "calculated response." In general, an estimated response is that in which a rigorous analytical process is avoided by relying on engineering judgment or simplistic mathematical approximations. Typically, estimated responses will be somewhat conservatively biased. In the context of SFR-B1, a typical example of an "estimated" response could be when an in-structure response of one building is used for another similar building. Another common example is when approximate RE ISRS are estimated by scaling design ISRS. SFR-B1 permits the use of estimated responses in CC-I for all SSCs, whereas for CC-II, calculated responses are required for risk-significant SSCs such that the approximations do not significantly bias the seismic PRA results or risk insights. In summary, SFR-B1 requires that the seismic response analysis be sufficiently realistic (or appropriately conservative) such that any approximations introduced do not significantly bias or alter the seismic PRA results or risk insights. A few key elements that should be considered are nonlinearity in response to increasing input levels, the definition of the site-specific input spectral shape and input level (e.g., RE), the reference parameter and control point, and the development of input time histories. This is not intended to be an exhaustive list since there are many more considerations in developing a realistic seismic response. SFR-B2 through SFR-B6 focus on several other elements of the seismic response analysis that are required to obtain a sufficiently realistic (or appropriately conservative) seismic response. See SFR-B1

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-21	The scaling procedures given in EPRI NP-6041 [S-7] and EPRI 3002012994 [S-4] may be used. Scaling of existing ISRS should consider the shapes of the original and new ground motion spectra, structural natural frequencies, mode shapes, and participation factors. Justification needs to be provided if there are significant differences in the phenomena that may be inadequately represented by linear scaling responses, including structural dynamic characteristics between the original model and the current configuration, foundation characteristics (e.g., non-linear soil properties), and ground motion spectra. See SFR-B2
S-N-22	The adequacy of building models for use in seismic PRA will depend on their ability to capture the realistic response of their as-built and as-operated condition. This represents a challenge for structural analysts since models should be developed as practical as possible to include those modeling features that will ensure a realistic seismic behavior. Important modeling features affecting seismic response include equipment masses, dynamic coupling of secondary systems, floor diaphragm flexibility, soil embedment, floor torsional effects, sloshing, directional coupling, rotational inertia, and torsional effects. Caution should be exercised when reusing older (e.g., design-basis) building models since important modeling details may have been defined with obsolete methods or conservative bias. Conservative biases would lead to a misrepresentation of the structural dynamic response. Structural modeling parameters with large sources of uncertainty should also be considered in a seismic PRA. These modeling parameters include, for example, structural damping and stiffness modifiers consistent with the response behaviors exhibited at the selected ground motion level as required in SFR-B1. Experience has shown that the effort to achieve realistic estimates of building modeling properties could require considerable analytical and computer time, thus incurring in excessive project resources. To this end, structural analysts should maintain a balanced philosophy between the interim objective of achieving reasonable estimates of median modeling features and the overall goal of achieving reliable PRA results for future risk-informed decisions. The structural modeling approach can be considered as a source of model uncertainty, and the effects of the modeling assumptions would need to be justified on the risk-informed decision being made. Assumed as-designed and as-intended-to-operate conditions are acceptable for reactors prior to construction. See SFR-B3
S-N-23	EPRI 3002012994 [S-4] provides guidance for determining median centered seismic response and its variability due to randomness and uncertainty in the various parameters affecting seismic response. Variability in the various parameters could also be estimated based on available test data with appropriate justification. SFR-B4 requires the estimation of the variability in the best-estimate response. The variability can be expressed as "composite" or separately as aleatory and epistemic. See SFR-B4

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-24	SSI effects could be significant for certain sites based on the site soil conditions, construction configuration, and/or structural properties. The potential effect of SSI, and the subsequent decision for whether SSI is considered or not, should be assessed and documented. If SSI effects are considered to be significant, ASCE-4 [S-3] and EPRI 3002012994 [S-4] provide guidelines for performing SSI analysis, including treatment of variabilities. Guidance is available [S-8] for conditions where SSI effects may not need to be considered, such as for rock sites. However, for rock sites, the effects of spatial incoherence of seismic ground motion could be significant and should be considered. Ground motion incoherence typically lowers the structural response at higher frequencies (>10Hz) while it typically increases rocking and torsion response. The distinction between CC-I and CC-II is twofold. First, CC-I does not specifically require SSI effects to be considered, whereas CC-II requires SSI effects to be considered if they are significant to structural response. Second, the level of rigor required when considering SSI effects is higher for CC-II than for CC-I. For example, for CC-I applications, a simplified stick model with soil springs may be sufficient to estimate SSI response, whereas for CC-II applications, a more detailed analysis accounting for the embedment effects, incoherency, etc., may be necessary to calculate SSI response. Additionally, for CC-I applications, it may be sufficient to use approximate soil properties based on a nearby facility or a site with similar materials (e.g., as long as they are reasonable and appropriate for the geotechnical materials and hazard level for the site in question), whereas CC-II applications requires the use of soil properties from site-specific geotechnical characterization with specific strain-compatibility to the hazard level(s) considered. Either deterministic or probabilistic SSI analyses can be performed. As soil properties (shear wave velocity, damping, etc.) are strain-dependent, soil properties for SSI should be consistent with the site seismic hazard level defined in SFR-B1. Variability in soil properties need to be considered. Whenever possible, it is preferred for variability in soil properties considered for SSI to be defined consistent with the variability in soil properties considered in site response analysis as part of probabilistic seismic hazard analysis. These variabilities should be propagated through the SSI analyses so that their effects on the structure response can be quantified. Quantification of the effect of variabilities can be reported in various forms, including as estimates of both median and some fractile (e.g., 84th percentile) of response, or as measures of structural response variability [e.g., COV, β_{RS}, ln(σ)] directly. See SFR-B5
S-N-25	This SR is not applicable to PRAs performed on a bounding site as the intent is for the fragility analysis to be decoupled from the hazard analysis to the extent practical. See SFR-B5

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-26	The probabilistic response analysis requires a sufficient number of simulations to be able to rigorously quantify the aleatory and epistemic variabilities in the free-field ground motion, the building and foundation media stiffness, damping values, etc. The treatment of the aleatory variabilities can be accomplished through the collection of ensembles of ground motion time history sets, preferably obtained from an earthquake catalog of recorded motions. Each of the input motion sets shall consist of two horizontal components and one vertical component. These time histories should be compatible with the seismic hazard (e.g., UHRS) at the appropriate control point. Epistemic variabilities quantify uncertainty in the behavior of the soil-structure system, such as uncertainty in soil shear modulus, soil material damping, and the structure dynamic characteristics. Uncertainty in the structural dynamic characteristics is typically addressed by varying the structure fixed-base frequencies and modal damping. The probability distribution function can be derived by various methods of sampling, including Latin Hypercube Sampling. The analyst must ensure that a sufficient number of simulations are performed to achieve stable probabilistic distributions for the response parameters. For example, a sensitivity study conducted in [S-21] demonstrated that the use of 30 Latin Hypercube Sampling was sufficient to yield stable median values and logarithmic standard deviations of selected response quantities. Additional guidance on the number of simulations is provided in ASCE-4 [S-3]. The probabilistic seismic response analyses performed in the early days would require generation of 30 sets of time histories for the input ground motions, which were defined by the median and 84th percentile ground response spectra (GRS). The 84th percentile GRS was used to account for uncertainty in the spectral shape (so-called "peak-to-valley variability"). The 30 sets of time histories were adjusted so that their median and 84th percentile ground response spectra would closely match the corresponding GRS. The other response variables explicitly considered in the probabilistic response analyses were structure stiffness and damping, and soil shear modulus and material damping. Thus, from the resulting statistically calculated median and 84th percentile ISRS, one could obtain a composite variability for response due to variability associated with the input ground motion, the structure model, and soils. Refer to Appendix H of EPRI Report 3002012994 [S-4] for historical context for why such peak-to-valley variability is no longer explicitly included in seismic response analysis. See SFR-B6

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-27	The term "inherently rugged" refers to seismic capacities well beyond the capacities of SSCs that normally govern seismic risk. As such, there is very high confidence inherently rugged SSCs will not significantly contribute to seismic risk, regardless of the site-specific seismic hazard level. Typical items include manual valves, check valves, and small, in-line strainers. The EPRI SPID [S-8] and EPRI Report 3002012994 [S-4] include extensive discussions on the meaning of inherently rugged, as well as a list of types of SSCs that are typically considered inherently rugged. SFR-C1 requires specifying the basis used for defining the list of what are considered as inherently rugged components, which, in general, should be regardless of the site seismicity. In practice, there could be scenarios where fragility analysts may want to expand the available generic lists of inherently rugged component types (such as provided in the EPRI Seismic PRA IG [S-11]) with the intent to screen a larger set of SSCs from inclusion in the systems model and risk quantification. For example, it is commonly acceptable to categorize manual valves as inherently rugged components without developing an explicit, rigorous justification for their seismic ruggedness. However, fragility analysts may judge other non-conventional SSC types (e.g., other than those listed in SPID [S-8]) as inherently rugged, such as small pumps, motor-operated valves, air-operated valves, or wall-mounted instruments. In this case, plant-specific justification should be provided to demonstrate that the additional SSCs identified as inherently rugged have sufficiently high capacities relative to other SEL SSCs to warrant screening them from the systems model and quantification. A similar example exists where certain SSCs have sufficiently low seismic demands (as opposed to sufficiently high capacities) relative to other SEL SSCs to warrant screening them from the systems model, such as when a portion of the plant is seismically isolated effectively reducing the seismic demand on SSCs supported "above" the isolators. See SFR-C1
S-N-28	The fragility analysis in a seismic PRA should focus project resources on SSCs that are important to plant risk. A target fragility level is first established by the systems analyst (as required in SPR-B5) in terms of a ground motion parameter, e.g., PGA or average spectral acceleration. The fragility analyst will then compare capacities of SEL SSCs (also expressed in terms of the reference ground motion parameter) to the target fragility level. SFR-C2 requires that the seismic PRA provide the basis and methodology employed for developing the methodology used to compare the SSC capacities to the target fragility level. Guidance that can be used for establishing the basis for target fragility is provided in various industry documents, (e.g., EPRI NP-6041 [S-7] and EPRI 3002012994 [S-4]) for developing SSC seismic capacities after satisfying specific caveats. For example, References [S-4] and [S-7] provide generic fragility screening-level seismic capacities as well as guidance on how to justify that SSCs meet the fragility screening levels. This can be used to satisfy SFR-C2 provided that the generic screening-level capacities are high enough to meet the target fragility established by the systems analyst (SPR-B5). However, the target fragility level (SPR-B5) in high-seismicity sites may be higher than the generic screening-level capacities provided in [S-4] and [S-7]. For these cases, the analyst should develop and justify alternate criteria to establish seismic capacities for comparison with the higher target fragility level (SPR-B5). This could be based on a combination of use of the site seismic-design criteria, site-specific test data, and bounding analyses. See SFR-C2
S-N-29	For PRAs performed during the pre-operational stage, this confirmation can be accomplished via interviews of knowledgeable personnel or review of design reports. See SFR-D1

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-30	It is important to note the difference between the intent of SFR-C2 and SFR-D1. First, SFR-C2 requires for the fragility analyst to clearly document the basis for screening components using industry-accepted methodologies, these being, for example, past experience data, engineering judgment, or conservative/generic capacity values. On the other hand, SFR-D1 is more SSC-specific in the sense that it requires a clear identification of those SSCs that meet the target fragility provided in SFR-C2. SFR-D1 also includes the requirement to ensure that anchorage or structural supporting condition of the component also meets the target fragility. This ensures that both functional and structural-related failure modes are assessed in the screening process. The target fragility screening criteria defined in HLR-SFR-C are applicable to SFR-D6, SFR-D7, and SFR-D8. See SFR-D1
S-N-31	Although a site-specific and plant-specific analysis of the fragilities of SSCs provides the most realistic estimates, for the pre-operational stage PRA, it may be acceptable to develop fragility estimates on some other basis (e.g., generic information), provided that the uncertainties introduced are acceptable for the application. See SFR-D2
S-N-32	Examples of investigations include walkdown(s), tabletop review, or computerized walkdown, as applicable. See SFR-D2
S-N-33	For PRAs performed during the pre-operational stage, it is assumed that seismic vulnerabilities are absent and the fragilities estimated based on design criteria or generic information are appropriate for PRA at the pre-operational stage. See SFR-D2
S-N-34	The purpose of the seismic PRA walkdown is to verify that the component fragility curves are consistent with the current plant configuration. The seismic PRA walkdown is vital to confirming screening applicability (SFR-D1), collecting information necessary for fragility calculations, and identifying anchorage and interaction concerns. Ideally, the walkdown team includes adequate experience to make appropriate judgments concerning credible failure mechanisms, potentially significant interactions, and information potentially significant to fragility calculations. See SFR-D2

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-35	In the context of fragility, a seismic vulnerability is defined as any failure mechanism for a SSC that could control the seismic capacity in the fragility analysis of that SSC. In addition to review of design documents, the identification of seismic vulnerabilities involves detailed plant investigations by engineers knowledgeable in seismic performance of SSCs, their design functions, and their critical failure mechanisms. Examples of seismic vulnerabilities include weaknesses in the anchorage load path, excess flexibility in the attachment and load path of internal sub-assemblies (which may lead to sensitivity to low frequency or vertical direction seismic input), insufficient commodity clearance, differential displacement issues, overhead seismic interaction falling hazards, and poor plant maintenance that may have an impact on component functionality. For CC-I, the focus is to identify seismic vulnerabilities so that the assumptions and the use of generic seismic fragilities are conservative. For example, if seismic-experience-based generic capacity is intended to be used for an air handling unit, then the investigation should ensure that all potential vulnerabilities that may result in capacity less than the generic capacity are identified. This includes not only verifying compliance with the applicable experience-based caveats and inclusion rules, but also verifying that all potential seismic interactions, such as potential falling of masonry walls, have capacities exceeding the seismic-experience-based capacity that will be assigned to that air handling unit. For CC-II, the focus is to identify seismic vulnerabilities so that the seismic fragility calculations can be realistic, as needed. It is critical that all seismic vulnerabilities that may control seismic fragility are captured during the investigations and carried through to the fragility analysis and that the identification of seismic vulnerabilities be thorough and realistic. At the time of the investigations, excess conservatism cannot be arbitrarily used because it is typically not known yet if the component is going to be a dominant contributor to plant risk. The investigations should realistically identify the seismic vulnerabilities appropriate for each SSC. For example, consider that an impact-sensitive component may be within close proximity to a poorly anchored heat exchanger. However, the configuration of floor penetrations associated with the heat exchanger and attached piping may preclude the heat exchanger from deflecting toward and reaching the component. In this case, the proximity to the heat exchanger is not a realistic seismic vulnerability for that component. If the fragility analysis were incorrectly governed by the capacity of the heat exchanger, then that low capacity fragility would not be realistic and may mask the seismic PRA from identifying true plant vulnerabilities. Conversely, as a second example, consider an electrical cabinet with seismic capacity verified by shake table fragility testing, in close proximity to a tall masonry wall with seismic capacity higher than the capacity of the cabinet. However, prior to experiencing seismic motion consistent with the seismic capacity level of the electrical cabinet, the masonry wall may deflect out of plane and strike the electrical cabinet with enough force such that functionality of the cabinet is lost. If the deflection of the masonry wall is not identified as a vulnerability during investigations, then the fragility analysis may significantly overestimate the seismic capacity of the electrical cabinet. See SFR-D3, SFR-D4
S-N-36	This SR is likely not applicable to PRAs performed during the pre-operational stage. See SFR-D3
S-N-37	This SR is not applicable to operating plants. See SFR-D4, SFR-E7, SFR-F3, SPR-E7, SPR-F4
S-N-38	Here, the term "failure mechanism" refers to the seismically induced failure of interest in a fragility calculation such as, for example, pull-out of anchors, excessive bending of a valve yoke, or circuit burnout in cabinets. See SFR-D5

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-39	During an investigation, the investigation team may observe hundreds of credible seismic interactions. For example, two adjacent conduits may impact each other during an earthquake. However, the earthquake experience data has shown that these types of interactions do not pose a risk to the intended plant safety functions. Thus, the investigation team must exercise judgment as to when a credible interaction may be risk-significant and warrant further evaluation. Guidance is available in EPRI 6041 [S-7] and the SQUG GIP [S-22]. Seismic interactions that may affect SSCs' intended functions or operator actions include proximity impacts (e.g., impact between cabinets), falling hazards (e.g., failure and falling of non-seismically designed SSCs and masonry walls), and differential displacements (e.g., differential building displacements). See SFR-D8
S-N-40	This SR requires the identification of relevant seismic-induced failure mechanisms of structures, equipment, and soil. These failure mechanisms become the focus for fragility calculations performed in SFR-E3. The failure mechanisms evaluated in the fragility calculations should be related to the credited SSC functions in the seismic PRA model. For example, equipment anchorage failure modes are evaluated because they can lead to functional failures in the equipment, and interaction failure modes such as block wall failures near SEL equipment should be evaluated if the interaction would prevent the equipment from performing the credited function. SFR-E1 involves the identification of relevant and realistic failure modes of structures, equipment, and soil. For structures, typical failure mechanisms include sliding, overturning, yielding, and excessive drift. For equipment, typical failure mechanisms include anchorage failure, functional failure, impact with adjacent equipment or structures, and bracing failures. For soils, typical failure mechanisms include liquefaction, slope instability, and excessive differential settlement. In CC-I, failure mechanisms can be identified in a less rigorous manner when compared with CC-II. This may take the form of identifying the most likely failure mechanism for a given SSC whereas in CC-II, it may be necessary to identify more than one likely failure mechanism to consider in the SSC fragility calculation. This SR allows the use of the conservative failure mechanism to establish the fragility parameters at the component level; however, the fragility analyst needs to have a proper understanding about the components' dependency. For example, the intent of the "rule of the box" for equipment is that all of the components mounted on or in this equipment are considered to be part of that equipment and do not have to be evaluated separately. The fragility analyst can identify the "rule of the box" components; however, auxiliary components that are not mounted on the equipment but are needed by the equipment to fulfill its intended function need to be evaluated separately. The fragility analyst can gather information about the dependency from the system engineers or by reviewing the plant drawings [e.g., system piping and instrumentation drawings (P&IDs), single line electrical, anchorage drawings, walkdown notes]. This may include sections of piping, cable trays, or supports that are not part of the failure mechanism but can impact the other components. Another example is the loss of off-site power (LOOP) with the typical generic fragility parameters, which allows to exclude the fragilities for components that are dependent on off-site power (OSP) at median 0.3 g earthquake or greater. However, fragilities for components that are dependent on OSP at lower than 0.3 g median earthquake need to be developed. The fragility analyst needs to identify and include any correlation of redundant components. Correlation may depend on component characteristics, physical separation, and location within the plant. The correlation, dependency, and failure mechanism may be used for combining SSCs into groups, which reduces the number of fragilities used in the plant response model. See SFR-E1, SFR-E2
S-N-41	This SR is not applicable to PRAs performed on a bounding site. See SFR-E2

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-42	Although a site-specific and plant-specific analysis of the fragilities of SSCs provides the most realistic estimates, for the pre-operational stage PRA, it may be acceptable to develop fragility estimates on some other basis (e.g., generic information), provided that the uncertainties introduced are acceptable for the application. For reactors in the pre-operational stage, decoupling the fragility analysis from the hazard analysis to the extent practical allows for the fragility analysis to bounding a range of sites. It may be necessary to use generic data for this purpose. See SFR-E3
S-N-43	Realistic and site-specific fragilities are required for the risk-significant SSCs in the seismic PRA model unless conservative or generic fragilities can be justified as being appropriate for the plant. The term "conservative" fragility refers to assumptions made in the fragility analysis which are purposely conservatively biased. For example, in a pump fragility calculation, the analyst may determine nozzle loads on the pump without crediting all of the supports on the attached piping, which, if credited, would increase the pump capacity. Justification for the conservatively biased pump fragility can be provided by reviewing in detail the dominant SSC contribution in the overall risk profile. SSCs that have a small impact on the risk profile may not require realistic fragilities. These small impact SSCs may be justified as appropriate for the plant through importance measures such as a low Fussell-Vesely value or by showing that further refinement in the fragility analysis would not appreciably change the SSC contribution. More detailed and realistic fragilities are required for SSCs that have a large impact on the overall risk profile if they cannot be justified as appropriate. Justification for these large-impact SSCs may include a sensitivity study that shows the result of an estimated higher capacity realistic fragility neither significantly changes the risk metrics nor the risk insights (e.g., does not create a masking concern). The combined effect of multiple generic or conservative fragilities should be considered in these sensitivity studies due to SSC dependency in the PRA model. The intent is to provide justification that no generic or conservative fragility is preventing an SSC from being identified as risk-significant (e.g., masking the contribution of other SSCs) in the seismic PRA model. It is understood that these sensitivity studies may result in reordering or shuffling of the top contributors or a single top contributing SSC could drop in importance with the remainder not substantially changed, and this would be acceptable. The masking concern would be a notable rise in risk significance (e.g., when a small contributor SSC instantly becomes a large contributor) during a sensitivity study on a large contributor SSC or group of SSCs. Some examples of generic fragilities that are often large contributors to a seismic PRA are LOOP and very small reactor coolant system boundary (RCB) breach. These and other generic fragilities may be appropriate for the plant given justification. Some generic fragilities like the very small RCB breach may provide a significant reduction in seismic event sequence frequencies when the median capacity is increased. However, a conservative value may still be acceptable if it is demonstrated that there is no masking effect. The term "failure mode" in SFR-E3 follows the same definition as in SFR-E1, that is, the seismically induced failure mechanism of interest in fragility calculations such as anchorage pull out, relay chatter, among many others. See SFR-E3

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-44	Functional failure of relays and other electro-mechanical contact devices is likely to occur under earthquake ground motions. Some of these functional failure modes may not affect the credited seismic PRA system functions (i.e., chatter acceptable) while others may lead to undesired system performance during an earthquake. For fragility analysis, the key analysis criterion is typically an assessment of a broadband capacity-to-demand comparison at the mounting point of the component over the frequency range of interest. Narrow banded demand and capacity peaks are typically clipped to determine the effective broadband capacity-to-demand evaluation. Relay and contactor seismic capacities are typically derived from shake table testing. For CC-I, estimates of parameters such as the in-structure response spectra, electrical cabinet natural frequencies, effective cabinet amplification, and representative component capacities can be used. The use of generic or conservative estimates should be justified in accordance with SFR-E4. For CC-II, the fragility calculations are expected to be more realistic and make use of plant-specific data. Parameters used in the fragility calculations should be median-centered without a conservative bias. The use of generic data should be justified in accordance with SFR-E4. For example, if more detailed fragility calculations for a relay or contactor would not result in a significant change in seismic event sequence frequencies, this can be used to demonstrate that the use of the generic or conservative fragility parameters are appropriate. See SFR-E4
S-N-45	It is expected that relay chatter will not play an important role in the safety response for non-LWRs. As such, relays may not be included in the SEL, making this SR not applicable. See SFR-E4
S-N-46	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See SFR-E7, SPR-E7
S-N-47	The purpose of this SR is to capture potentially significant assumptions that may impact the quantification results (see SPR-E8). These are different from parametric uncertainties of variables affecting the fragility value of components that are already accounted for in the fragility analysis. In addition to identifying the potentially significant assumptions, an estimate of change in the fragility values of the affected components needs to be made so that the impact on the quantification results could be determined. Examples of potentially significant assumptions in fragility analysis include, but are not limited to, the following: *(a)* use of representative or conservative fragility values for risk-significant components (see SFR-E3 above); *(b)* use of generic seismic experience data in lieu of plant-specific seismic qualification test data for components; *(c)* lumped mass spring models in lieu of 3D finite element models in the structural response analysis; *(d)* neglecting the effects of structure-soil-structure interaction; *(e)* neglecting the effects of Ground Motion Incoherence. EPRI Technical Update 1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty" [RI-9] expands on the above assumptions and provides more details for seismic PRA applications. See SFR-E7

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-48	The documentation of the fragility results needs to provide the required information such that the results obtained can be followed and replicated, if needed, in future PRA upgrades. A systematic process should be used in referencing different sources for information used in the analysis and calculations. The methodology used to perform the building seismic response and fragility analyses needs to be described in the documentation so that it could facilitate the peer review process and for PRA applications. A thorough documentation of engineering judgment needs to be included to facilitate peer review. See SFR-F1
S-N-49	Sources of model uncertainty are documented and their impact on the model needs to be evaluated. An example for the source of model uncertainty is an issue for which there is no consensus in approach (e.g., frequency range of interest, in-cabinet amplification factor used in relay fragilities, degree of cracking in buildings) and where the approach is known to have impact on the fragility analysis. See SFR-F2
S-N-50	The intent of this requirement is to ensure the entire spectrum of seismically induced initiators is systematically evaluated, ranging from large catastrophic events resulting in major structural collapse to smaller magnitude events possibly resulting in a manual or automatic trip due to the seismic event being above operational limits. The requirement also focuses the attention of the analyst on combined events such as LOOP and/or an RCB breach coincident with other initiators that are normally not considered in the initiating event categorization used in the internal events PRAs. The systematic identification process is intended to allow, where applicable, screening or subsuming. See SPR-A1
S-N-51	Attention should be given to secondary events such as seismically induced fires, internal and external floods, and other similar events, as applicable. Existing guidance (see for example, [S-11]) provides a reasonably complete list of seismically induced external hazards to be addressed for the possibility of seismically induced events. As far as seismically induced internal floods and internal fires, the flood sources and fire ignition sources identified as part of the Internal Flood PRAs and Internal Fire PRAs are, if available, an appropriate and consistent starting point. Note, finally, that this requirement works in conjunction with Requirements SHA-H1 and SHA-H2 in the identification of other non-vibratory hazards generated by the seismic event (e.g., soil liquefaction, fault displacement) with emphasis on the effect on the plant. In principle, any hazard that does not screen out from SHA-H1 and SHA-H2 needs to be picked up in the scope of the Seismic PRA explicit modeling. The systematic identification process is intended to allow, where applicable, screening or subsuming. See SPR-A2
S-N-52	The internal events PRA model may or may not include event sequences involving two or more reactors or radionuclide sources. However, if seismically induced failures impact two or more reactors or sources, such event sequences may be introduced for treatment of this hazard group. See SPR-B1, SPR-C1
S-N-53	It has been shown that even minor unaddressed or insufficient resolution of findings in the internal events model can result in significantly amplified errors in the seismic model. Therefore, care should be taken to look for these cascading effects in the seismic model. See SPR-B2

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-54	The target fragility represents a threshold in seismic capacity that corresponds, when integrated with the site-specific hazard, to an event that is less than risk-significant and as such may be omitted from explicit modeling in the seismic PRA. A target (or threshold) fragility addresses potentially a large number of components (even when used for a single one) in the plant (tens or even hundreds of components) and therefore the numerical value associated for the cumulative screening. The target fragility may be different for different event sequence families and are to be defined independently from other screening considerations, if used, and from correlation groups and component grouping of fragilities. Reference [S-8] established the 5% criteria as acceptable. While a target fragility is likely selected early in the development of the seismic PRA, to aid in the planning and execution of the fragility analysis effort, this SR is to be addressed in the context of the finalized seismic PRA. See SPR-B5
S-N-55	Some non-LWRs are expected to exclude relays from either their plant-design safety case, thus meeting the intent of this SR by design. See SPR-B6
S-N-56	This SR is not applicable to PRAs performed for a single reactor plant. See SPR-B13
S-N-57	In practice, the SSCs included in the SEL are accompanied with essential details such as failure mode of interest, building location of component, component description, among several others. Reference [S-11] provides guidance on the details typically included in the SEL. See SPR-C1

Table S-1 Notes Supporting Seismic PRA Requirements (Cont'd)

Number	Notes
S-N-58	Typical examples of failure modes of interest in a seismic PRA may include failure of a valve to open on demand, loss of function during an earthquake, or rupture of a pressure boundary. Note that these failure modes are defined by the systems analyst and may not be the same as that defined by the fragility analyst. In practice, fragility analysts will identify the failure mode of a component based on vulnerabilities identified during the investigations (see SFR-D2) or the most likely lower bound seismically induced failure mechanism. This identification is typically based on experience, available test data or analytical procedures. Hence, the continuous interaction between systems and fragility analysts when defining the failure mode represented in a fragility curve and that credited in the model is important for identifying failure modes. It is also worth noting that what SFR-E1 refers to as the "relevant" failure mechanism corresponds to the failure mode defined here in SPR-C6 by the systems analyst, say, "failure to close" or "fail during earthquake." Once this "relevant" failure mechanism has been clearly established, then the fragility analyst will proceed to assess which seismically induced failure could most likely lead to the failure mode defined here in SPR-C6. It may be possible that the systems analyst may be interested in the consequences rather than a seismically induced failure mode. An example for this case could be the failure definition of a motor control center (MCC). In one case, failure of the MCC may lead to adverse changes of state in the plant, and failure would be defined as loss of function during the seismic event. In another case, failure of the MCC during a seismic event may be acceptable, but after the event, the MCC should function. Fragility analysts will derive two distinctly different capacities, i.e., "function during" or "function after." Another example for distinguishing consequences from failure modes is in the case of failure of heat exchangers. The fragility curve for a heat exchanger may be derived from that for failure of anchors. However, the consequences modeled in the seismic PRA model may be related to flooding of the area. Such a scenario indicates that there is still a significant margin between the failure mechanism defined by the fragility analyst and the failure mode credited in the seismic PRA model. This scenario is an example of a source of considerable conservatism since the median capacity based on failure of the anchors will grossly underestimate the seismic capacity corresponding to the failure mechanism, leading to flooding of the area. In most cases, precluding a more detailed analysis, the progressive failure mechanism rather than pullout of an anchor should be used to judge the overall contribution of such a component of plant risk. See SPR-C6
S-N-59	Examples of seismic hazard effects on the control room and ex-control room human actions include the following: *(a)* training and procedures; *(b)* additional workload and stress; *(c)* effects of the seismic event on mitigation; *(d)* required response, timing, and accessibility; *(e)* potential for physical harm; and *(f)* seismic-specific job aids and training. See SPR-D5
S-N-60	Caution: When satisfying Requirement ESQ-B3, the 5% truncation rule noted in the requirement is viewed to only be an example and is not intended to be a requirement. See SPR-E4

S.2 EXPLANATORY MATERIAL ASSOCIATED WITH SEISMIC PRA

This Standard is anticipated to be used by non-LWR reactor designers and vendors prior to site selection (i.e., at the time of design certification application). The Seismic PRA as well as external flooding, high winds, and other external hazard technical elements and sub-elements, will require a site-specific analysis (e.g., probabilistic seismic hazard analysis). Prior to site selection, designers may seek to perform a seismic analysis of their design. The following guidelines are provided to aid designers and vendors in identifying a bounding set of analyses for Seismic PRA and external hazards.

(a) The designer or vendor will select the design-basis external hazards (seismic, tornado, etc.) that generally envelop the potential sites where the plant would be located. For seismic events, seismic response analysis is done for different site conditions (e.g., soil profile) to obtain a bounding set of responses.

(b) At the time of the design certification, the fragilities of some SSCs may need to be estimated using generic information and design criteria. The seismic hazard (in terms of hazard curve and ground response spectra) is chosen to envelop the potential sites. The goal of a seismic PRA at the design certification application stage is to identify vulnerabilities and risk insights associated with the design. The PRA at this stage also assures the designer that the plant would meet the associated risk criteria when complete. Even without site-specific information, the seismic PRA should reveal any unique seismically induced event sequences and event sequence families that could be efficiently addressed during the design. If the seismically induced event sequences do not fit into existing release categories, new release categories are defined for which new mechanistic source terms and radiological consequences are needed. This will facilitate the inclusion of seismically induced event sequences into the Risk Integration.

(c) After the design certification application, a site "A" is chosen and the detailed design of the SSCs completed (or checked) using site-specific information (e.g., soil profile). The hazard curve for the site is used in the quantification along with the plant- and -design-specific fragilities of SSCs. The seismically induced risk is re-evaluated to capture the introduction of a site-specific hazard analysis, and site- and design-dependent fragility analysis and is accomplished using the same process as was used for the design certification.

When a second site "B" is selected, the designer is expected to verify that the site "A" chosen after the design-certification application is suitable at site "B"; any needed modification resulting from site characteristics at B will have to be made. Similarly, the seismic PRA will be modified to represent the site-specific conditions at "B" and the hazard curve for site "B." The SRs that address site-specific information and conditions when a site has been selected should be applied.

S.2.1 Commentary on Seismic Hazard Analysis

Seismic Hazard Analysis provides both an assessment of vibratory motion using a probabilistic seismic hazard analysis and an assessment of non-vibratory hazards. The results of the probabilistic seismic hazard analysis are usually expressed in terms of the frequency distribution of the peak value of a series of horizontal ground motion parameters (e.g., PGA) over a range of specified time intervals. As described in References [S-13] and [S-14], steps of this analysis are typically broken into three areas: (a) seismic source characterization, (b) GMC, and (c) site response. Probabilistic seismic hazard analysis is a site-specific analysis that may or may not build on available regional studies. If a regional seismic source characterization or GMC study is used as a starting point, the requirements related to use of an existing study apply. The results of a probabilistic seismic hazard analysis can also be used to develop uniform hazard spectra (also known as response spectra). Seismic hazard analyses also include assessment of vertical ground motion, which is typically calculated using the horizontal ground motion coupled with a V/H ratio.

Currently, the state of practice for probabilistic seismic hazard analysis model development follows guidelines developed by the SSHAC, as described in References [S-13] and [S-14] (hereafter noted as the "SSHAC guidelines"). The SSHAC guidelines describe a structured approach of probabilistic seismic hazard analysis model development that addresses most of the requirements related to HLR-SHA-A through HLR-SHA-D and HLR-SHA-F. The SSHAC guidelines have significant documentation requirements consistent with HLR-SHA-I. The SSHAC guidelines define four levels of study, with each study increasing in complexity. Levels 3 and 4, which are equally acceptable for developing probabilistic seismic hazard analysis appropriate for nuclear applications, provide the highest level of assurance that the objectives of the SSHAC process are met. Level 2 studies can be used to update or refine regional studies for site-specific use, as described in [S-13]. Level 1 and 2 studies also provide the basis for an assessment of existing studies. Regardless of the SSHAC study level, the objective of the SSHAC process is the same—to develop a model that represents the center, body, and range of the technically defensible interpretations of the available earth science information.

In probabilistic seismic hazard analysis, two different classes of uncertainties are identified and addressed throughout the process. Lack-of-knowledge uncertainties or epistemic uncertainties arise from imperfect scientific understanding that can, in principle, be reduced through additional research and acquisition of data. Epistemic uncertainties are often addressed in the modeling process through the use of logic trees or the use of sensitivity studies, which provide a quantifiable and transparent approach and lead to a family of hazard curves. The aleatory or random uncertainties (often called aleatory variability) are those uncertainties that, for all practical purposes, cannot be known in detail or cannot be reduced. Aleatory variability is typically addressed through the use of parameter distributions.

These two classes of uncertainties should be identified, quantified, and tracked separately throughout the probabilistic seismic hazard analysis process to the extent possible. Although some applications may use the mean hazard curve that includes combined uncertainties instead of the complete family of hazard curves, (see NUREG-1407 [S-16] for examples), maintaining the distinction in the nature of uncertainties is crucial for the development of the probabilistic seismic hazard analysis and is useful for identification of vulnerabilities and ranking dominant sequences and contributors. In probabilistic seismic hazard analysis, this distinction is maintained to understand and communicate the sources of uncertainties throughout the process.

Site response analyses are performed to quantify how near-surface geologic materials and their dynamic properties modify seismic vibratory motions entering the site from the underlying rock. In the past, site response analysis was performed as a separate calculation using as input the results from a rock-based probabilistic seismic hazard analysis. However, recently, it has been more common for site response to be directly incorporated into the probabilistic seismic hazard analysis integral using method 3 of [S-23]. If it is not directly incorporated into the probabilistic seismic hazard analysis, the soil amplification functions from the site response are applied to the uniform hazard spectra. In current practice, the site response aspects of Seismic Hazard Analysis are not subject to all the requirements of the SSHAC process although the most important aspects of the SSHAC process are increasingly applied. As with other elements of Seismic Hazard Analysis, the identification, quantification, and tracking of uncertainties are key components of site response evaluation activities.

The hazard estimates depend on uncertain estimates of ground motion propagation, upper-bound magnitudes, and the geometry of the postulated seismic sources, as well as on numerical treatment of source boundaries. Such uncertainties are included in the hazard analysis through the use of parameter distributions and logic trees. Parameter distributions are used to quantify aleatory uncertainties. Epistemic uncertainties are documented and quantified through the use of logic trees with probabilities assigned to alternative data, models, and methods. The annual frequencies of exceeding specified values of the ground motion parameters are displayed as a family of curves with different confidence levels.

For further details on seismic hazard analysis methods, the reader is referred to References [S-13] and [S-14]. An example case study of a probabilistic seismic hazard analysis using this guidance is found in Reference [S-23]. Typical results of a probabilistic seismic hazard analysis include families of seismic hazard curves in terms of PGA or spectral acceleration values at different frequencies and site-specific ground motion response spectra.

Table S-2 High Level Requirements for Probabilistic Seismic Hazard Analysis

HLR	Commentary
HLR-SHA-A	The SSHAC guidelines describe a structured process for assessing seismic vibratory ground motion on both a site and a regional level. Site-specific probabilistic seismic hazard analysis that include site-specific site response analysis, if appropriate, should be performed consistent with the SSHAC guidelines. The SSHAC level used for the study should be consistent with the guidelines provided in [S-13] and [S-14].
HLR-SHA-B	The SSHAC process begins with a comprehensive effort to identify, collect, and evaluate all available data, models, and methods for probabilistic seismic hazard analysis model development. The effort should follow guidance in [S-13]. Data and information relevant to site-specific analysis needs to be collected. NRC has provided guidance for data collection expected for licensing of new reactors in [S-19] and [S-20].
HLR-SHA-C	The SSHAC guidelines [S-13] provide guidance on seismic source characterization modeling for input to probabilistic seismic hazard analysis evaluations. An example of a regional study conducted using the SSHAC Level 3 approach is documented in [S-18].
HLR-SHA-D	The SSHAC guidelines [S-13] provide guidance on ground motion characterization modeling for input to probabilistic seismic hazard analysis evaluations.
HLR-SHA-E	Approaches for incorporating site response in the probabilistic seismic hazard analysis are provided in [S-23], with method 3 being preferred, particularly for new plants.
HLR-SHA-F	A mean estimate of the frequency of exceedance at any PGA and other spectral frequencies is calculated based on the weighted sum of the frequencies of exceedance at this acceleration given by the different hazard curves; the weighting factor is the probability assigned to the branch of the logic tree. Thus, the probabilistic seismic hazard analysis embeds uncertainties in the core of the methodology, and results are expressed in terms of likelihood—estimated probabilities in a given time period or estimated frequencies—that ground motions of various amplitudes will occur at a given site. Uncertainties must be carried through the site response analysis.
HLR-SHA-G	The spectral shape should be determined using the most risk-significant annual probability of exceedance. At times, the spectral shape at the design level is used as a starting point. In these cases, the shape of the response spectra at the most risk-significant annual probability of exceedance should be determined and checked against the design-spectral shape.

Table S-2 High Level Requirements for Probabilistic Seismic Hazard Analysis

<table>
<tr><th>HLR</th><th>Commentary</th></tr>
<tr><td>HLR-SHA-H</td><td>Non-vibratory seismic hazards are addressed through a three-step process that begins with screening for the hazard based on the potential for the hazard to occur at the site (not the potential impact of the hazard). If the hazard cannot be screened out, an analysis is performed in step two to determine the probability of hazard levels appropriate for input to the fragility and plant response evaluations. In step three of the process, the potential impact of the non-vibratory hazard is determined.</td></tr>
<tr><td>HLR-SHA-I</td><td>
<table>
<tr><th>SHA</th><th>SFR</th><th>SPR</th><th>Notes</th></tr>
<tr><td>SHA-H1</td><td></td><td></td><td>Can the secondary hazard be entirely screened from further evaluation based on the hazard and the site (e.g. liquefaction on a rock site, very low probabilities of liquefaction for very low annual probabilities of exceedance ground motions, or no potential fault displacement phenomena in the cite vicinity area.</td></tr>
<tr><td></td><td></td><td>SPR-A2</td><td>Identify Seismically-induced initiating events due to secondary hazards that do not screen.</td></tr>
<tr><td></td><td></td><td>SPR-C5/ SPR-C6</td><td>Define SEL including SSC's identified in SPR-A2</td></tr>
<tr><td></td><td>SHA-H2 Screening</td><td></td><td>SHA-H2 Screening type 1 - remove SEL items that cannot be impacted due to location.</td></tr>
<tr><td></td><td>SFR-E2</td><td></td><td>Determine the parameters of interest for the appropriate failure mechanisms for the SEL items that may be impacted.</td></tr>
<tr><td>SHA-H2</td><td></td><td></td><td>Determine the frequency of level of the hazard parameters parameters of interest identified in SFR-E2</td></tr>
<tr><td></td><td>SFR-E2</td><td></td><td>Determine the fragility for the SSCs based on the parameters of interest identified in SFR-E2</td></tr>
<tr><td></td><td>SHA-H2 Screening*</td><td></td><td>SHA-H2 screening type 2 - Remove failure of SEL items from secondary hazards from consideration based on the failure frequency using the hazard curve and the fragility curve.</td></tr>
<tr><td></td><td></td><td>HLR-SPR-E</td><td>Quantification</td></tr>
<tr><td colspan="4">*Once the fragility and hazard curves are developed, the information could be used directly in the quantification, rather than applying a second screening evaluation. This would eliminate the need for justification for "pruning" the model.</td></tr>
</table>
</td></tr>
</table>

S.2.2 Commentary for Seismic Fragility Analysis

Fragility curves in a seismic PRA should capture the realistic seismic behavior of, SSCs under a range of ground motion intensity levels, without having either conservative or optimistic bias. This principle is consistent with the intent that a seismic PRA realistically estimates the seismic response of plant systems against a range of seismic scenarios. This response is affected by both shape and median-centered parameters of the fragility curve. In other words, a conservative or unrealistic estimation of a median-centered ground motion parameter and variability in a fragility curve could mask individual SSCs that dominate seismic event sequence frequencies and therefore lead to unreliable PRA insights.

Fragility curves are derived using the probability density functions of the seismic demand and capacity parameters. The log-normal function has been generally used to model the random variables related to a component's capacity and ground-motion intensity, such as PGA or spectral acceleration. Even though other probability density functions can be used, the log-normal distribution has properties that facilitate the fragility analysis. Usually, the fragility curve is constructed by estimating a median ground motion acceleration Am and logarithmic standard deviations for uncertainty β_U and randomness β_R.

The practice to develop fragilities in seismic PRA for nuclear plants has been centered around six (6) interrelated areas: (a) definition of scope of fragility analysis, (b) building response analysis, (c) screening of SSCs, (d) plant walkdowns, (e) estimation of seismic fragility parameters, and (f) the documentation of the fragility analysis. The SRs for the Seismic Fragility Analysis Section of this Standard are organized around these six (6) core areas.

The comments offered herein aim to provide a background on the theory and practice that contributed to developing each requirement. In addition, practical scenarios are provided to enhance the understanding and the effort required for compliance with the requirements.

Table S-3 High Level Requirements for Seismic Fragility Analysis

HLR	Commentary
HLR-SFR-A	The scope of the Seismic Fragility Analysis is typically defined in the form of a SEL. This list generally includes identification of the SSCs that are credited in a seismic PRA and their descriptions, building locations (building, floor, and room number), failure modes of interest, and plant systems.
HLR-SFR-B	This requirement addresses the need to provide seismic response parameters such as displacements and in-structure accelerations that represent a realistic estimate of failure level of SSCs. This is of utmost importance in a seismic PRA since strong engineering judgment should be exercised to define the appropriate level of ground input to justify failure of SSCs across a sweep of seismic initiating events. This needs to be done in accordance with site-specific hazard and plant systems response characteristics.
HLR-SFR-C	Screening allows the analyst to focus resources on areas in a seismic PRA that drive the plant risk levels.
HLR-SFR-D	Fragilities of SSCs in a seismic PRA should represent as-built and as-operated conditions. The general practice to achieve this requirement is through plant walkdowns. For more than 40 years, experience has shown that plant walkdowns provide the seismic PRA team with the practical sense of seismic ruggedness in the plant as well as the identification of credible seismic-induced failure mode(s) of the SSCs often missed from design data, i.e., drawings or computer visualization models.
HLR-SFR-E	This requirement focuses on the mathematical approach used to establish the parameters defining the seismic fragility curve of SSCs. The fragility analyst should ensure that the variability and median values associated with variables affecting capacity and demand of SSCs are representative of the seismic-induced failure directly leading to the failure mode of importance to the seismic PRA.
HLR-SFR-F	Although the seismic PRA is developed as a snapshot in time, its use is intended for future risk-informed programs by several users across different fields of expertise. Hence, it is imperative to document the fragility analysis in a manner that facilitates peer reviews and future updates/upgrades.

S.2.3 Commentary for Seismic Plant Response Model

In general, the Seismic Plant Response Model is developed from the internal events PRA by first reviewing plant safety systems from the perspective of seismic safety and subsequently modifying the event and fault trees according to the seismic-specific initiating events. Among the characteristics of a seismic PRA model are the inclusion of the entire range of postulated potential earthquake ground motion levels; considering that seismic events may damage passive SSCs typically not modeled in internal event PRAs; that seismic events may simultaneously damage multiple redundant SSCs, thus requiring a combination of plant system responses; and consistent propagation of large uncertainties in the seismic hazard and fragility to produce the confidence ranges on seismic event sequence frequencies.

In recent years, significant advances in methodology for systems modeling and quantification in a seismic PRA have surfaced mainly in part to the insights from the seismic PRAs in the U.S. in response to 50.54 (f) letter [S-1]. Among these advances are the considerable progress made in areas such as seismic-induced fires and flooding, modeling of human response actions, and correlation between seismic failures. Significant progress has also been made toward a more integrated and collaborative effort between hazard, fragility, and systems analysts.

The requirements in Seismic Plant Response Model were revised with the intent to incorporate these advances in technology resulting from recent seismic PRA experiences in the U.S. The commentary notes provided here are intended to clarify the intent of the requirements as well as to facilitate collaboration among other technical elements, i.e., hazard (Seismic Hazard Analysis) and fragility (Seismic Fragility Analysis).

A general methodology for the modeling and quantification of a seismic PRA is documented in references such as EPRI-3002000709 [S-11], EPRI-1020756 [S-21], and EPRI-1025294 [S-10].

4.3.11 Hazards Screening Analysis (HS)

This Section presents the technical requirements associated with Hazards Screening Analysis. These screening criteria may not apply if not screened out earlier. See the High Winds PRA and External Flooding PRA sections for screening of high wind and external flood hazard or hazard groups.

4.3.11.1 Objectives and Technical Requirements for Hazards Screening Analysis

The objectives of the Hazards Screening Analysis ensure that

(a) potential hazards that may affect the nuclear power plant are identified;

(b) a set of qualitative screening criteria is specified;

(c) a demonstrably conservative analysis is used when quantitively screening out a hazard;

(d) an investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable] is conducted to establish or confirm as-built, as-operated or as-designed, as-intended-to-operate conditions;

(e) the Hazards Screening Analysis is documented to provide traceability of the work.

Table 4.3.11.1-1 High Level Requirements for Hazards Screening Analysis

Designator	Requirement
HLR-HS-A	Potential hazards that may affect the plant shall be identified for all sources of radioactive material and plant operating states within the scope of the PRA.
HLR-HS-B	Preliminary screening, if performed, shall be performed using a defined set of screening criteria.
HLR-HS-C	A demonstrably conservative analysis, if used for screening, shall be performed using defined quantitative screening criteria.
HLR-HS-D	The hazard screening evaluation shall incorporate the data and findings of investigation(s) of the plant (and its surroundings, as applicable to the hazard or hazard group) to establish or confirm either the as-built, as-operated or as-designed, as-intended-to-operate conditions.
HLR-HS-E	Documentation of the Hazards Screening Analysis shall provide traceability of the work.

Table 4.3.11.1-2 Supporting Requirements for HLR-HS-A

Potential hazards that may affect the plant shall be identified for all sources of radioactive material and plant operating states within the scope of the PRA. (HLR-HS-A)

Index No. HS-A	Capability Category I	Capability Category II
HS-A1	For the Hazards Screening Analysis, either *(a)* IDENTIFY the site at which the reactor being analyzed is located; or *(b)* DESCRIBE a bounding site and JUSTIFY that the bounding site bounds the list of sites in the scope of the PRA. See Note HS-N-1	
HS-A2	IDENTIFY hazards and hazard groups across all sources of radioactive material and plant operating states within the scope of the PRA that include those enumerated in industry guidelines and examined in past studies. See Note HS-N-2	
HS-A3	IDENTIFY site- and design-unique hazards and hazard groups not already identified in Requirement HS-A2. See Note HS-N-3, HS-N-4	
HS-A4	IDENTIFY secondary hazards associated with hazards and hazard groups from Requirements HS-A2 and HS-A3. See Note HS-N-4, HS-N-5, HS-N-6	

Table 4.3.11.1-3 Supporting Requirements for HLR-HS-B

Preliminary screening, if performed, shall be performed using a defined set of screening criteria. (HLR-HS-B)

Index No. HS-B	Capability Category I	Capability Category II
HS-B1	REVIEW information about the plant's design and either licensing basis or safety case relevant to each hazard. See Note HS-N-7	
HS-B2	For PRAs performed on operating plants, REVIEW information about regional-, industrial-, governmental-, and plant-funded evaluations for each hazard, if available. See Note HS-N-6, HS-N-7	
HS-B3	For operating plants, REVIEW major changes or updates since the design or operating license was issued as applicable. In particular, as germane to the given, REVIEW all of the following: *(a)* military and industrial facilities in proximity of the site or bounding site; *(b)* on-site storage or other activities involving hazardous materials; *(c)* nearby transportation; *(d)* nearby pipelines; *(e)* air routes; and *(f)* any other on-site or off-site changes that could affect the original design conditions. See Note HS-N-6, HS-N-8	
HS-B4	INCLUDE consideration of secondary hazard(s) or hazard group(s) in the qualitative screening process for hazards or hazard groups.	
HS-B5	USE SCR-3 in Table 1.10-1 when screening out a hazard or hazard group by showing that either *(a)* the hazard or hazard group cannot physically impact the plant or plant operations (e.g., it cannot occur close enough to the plant to affect it); *(b)* the hazard or hazard group does not result in a plant trip (manual or automatic) or require a plant shutdown; *(c)* the hazard or hazard group is included in the definition of another hazard; *(d)* the hazard or hazard group could not result in worse effects to the plant as another hazard that has a significantly higher frequency; *(e)* the hazard or hazard group is slow in developing, and there is demonstrably sufficient time to eliminate the source of the threat or to provide an adequate response; *(f)* the hazard or hazard group cannot produce a consequence above the value set in RI-A5.	
HS-B6	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with qualitative hazard screening in a manner that supports the Supporting Requirements (SRs) of HLR-ESQ-E and HLR-RCQ-C.	
HS-B7	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built and as-operated details that influence qualitative hazard screening. See Note HS-N-9, HS-N-10	

Table 4.3.11.1-4 Supporting Requirements for HLR-HS-C

A demonstrably conservative analysis, if used for screening, shall be performed using defined quantitative screening criteria. (HLR-HS-C)

Index No. HS-C	Capability Category I	Capability Category II
HS-C1	CALCULATE either the mean or demonstrably conservative frequency of occurrence or exceedance (as applicable) and associated parameters (e.g., loading magnitudes) of the hazards not qualitatively screened out in HLR-HS-B.	
HS-C2	USE applicable databases and information or JUSTIFY an alternative.	
HS-C3	IDENTIFY those structures, systems, and components (SSCs), and associated failure modes, required to maintain the plant in operation or that are required to respond to an initiating event to prevent releases of radioactive material, that are vulnerable to the hazard.	

Table 4.3.11.1-4 Supporting Requirements for HLR-HS-C (Cont'd)

A demonstrably conservative analysis, if used for screening, shall be performed using defined quantitative screening criteria. (HLR-HS-C)

Index No. HS-C	Capability Category I / Capability Category II
HS-C4	USE the event sequences and the systems logic model from internal event PRA models as the basis of the hazard screening plant response model. INCLUDE additional event sequences, as applicable, associated with hazard-induced initiating events that are not included in the internal events PRA model such as events that impact two or more reactors or sources of radioactive material. See IE-A16 See Note HS-N-11, HS-N-12
HS-C5	ENSURE that the findings from peer reviews identifying exceptions and deficiencies for the internal events PRA are dispositioned, or that they do not adversely affect the development of the hazard screening plant-response model. See Note HS-N-13
HS-C6	CALCULATE a demonstrably conservative conditional failure probability of mitigating functions using the internal events plant response model or USE a conditional failure probability of mitigating functions of unity (1.0). If plant response modeling is performed, SATISFY SRs of HLR-SY-A and HLR-SY-B for Systems Analysis, except where the requirements are not applicable. See Note HS-N-14
HS-C7	CALCULATE demonstrably conservative event sequence and event sequence family frequencies for the hazard by doing the following: *(a)* for discrete hazard, the product of the hazard frequency, and conditional failure probability of mitigating functions, as calculated in HS-C6; or *(b)* for hazard characterized by hazard curve, dividing the hazard curve into hazard intervals and summing for all intervals the product of the hazard interval frequency and associated interval conditional failure probability of mitigating functions (fragility analysis), as calculated in HS-C6; or *(c)* including the hazard induced initiating events and the systems or functions assumed rendered unavailable by the hazard into the internal events PRA model; or *(d)* using a hazard-specific model, as appropriate, using demonstrably conservative assessments of the impact of the hazard or hazard group on mitigating functions (fragility analysis). See Note HS-N-15
HS-C8	ADDRESS secondary hazard(s) in the hazard quantitative screening process.
HS-C9	When human actions are credited in the screening evaluation, ENSURE that hazard-induced impacts on human error probabilities are included as applicable.
HS-C10	IDENTIFY the consequences for the event sequence families impacted by the hazard as prescribed in Requirement RCQ-A3. See Note HS-N-16
HS-C11	For quantitatively screening out a given hazard or hazard group other than internal events or seismic events, USE the requirements of SCR-1 or SCR-2 in Table 1.10-1 or JUSTIFY an alternative criterion.
HS-C12	RETAIN the results of the screening model in the PRA. See Note HS-N-17
HS-C13	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with quantitative hazard screening in a manner that supports Requirement RI-C1.
HS-C14	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built and as-operated details that influence quantitative hazard screening. See Note HS-N-9, HS-N-10

Table 4.3.11.1-5 Supporting Requirements for HLR-HS-D

The hazard screening evaluation shall incorporate the data and findings of investigation(s) of the plant (and its surroundings, as applicable to the hazard or hazard group) to establish or confirm either the as-built, as-operated or as-designed, as-intended-to-operate conditions. (HLR-HS-D)

Index No. HS-D	Capability Category I	Capability Category II
HS-D1	CONFIRM that the basis for the screening out of a hazard or hazard group represents either the as-built, as-operated or as-designed, as-intended-to-operate, as applicable, conditions through data and findings of investigation(s) of the plant (and its surroundings, as applicable to the hazard). See Note HS-N-18, HS-N-19	

Table 4.3.11.1-6 Supporting Requirements for HLR-HS-E

Documentation of the Hazards Screening Analysis shall provide traceability of the work. (HLR-HS-E)

Index No. HS-E	Capability Category I	Capability Category II
HS-E1	DOCUMENT the process used in the Hazards Screening Analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* list of hazards addressed in the analysis and which hazards were screened out from further detailed analyses; *(b)* approach used for the screening (qualitative screening or demonstrably conservative analysis) and the screening criteria used for each hazard or hazard group that is screened out; *(c)* engineering or other analysis performed to support the screening out of a hazard or hazard group or in the demonstrably conservative assessment of a hazard or hazard group; *(d)* event sequence family frequency and mapped consequence results from quantitative screening calculations.	
HS-E2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement HS-C13) associated with the Hazards Screening Analysis.	
HS-E3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Hazards Screening Analysis. See HS-C14 See Note HS-N-10	
HS-E4	For Hazards Screening Analysis PRAs using a bounding site, DOCUMENT the basis for the selection of the bounding site characteristics that bound the range of sites for which the plant is designed and the justification of the applicability of the bounding site. See Note HS-N-20	

4.3.11.2 Peer Review Requirements for Hazards Screening Analysis

4.3.11.2.1 Purpose

This Section states requirements for peer review of screening and conservative analyses of hazards.

4.3.11.2.2 Peer Review Team Composition and Personnel Qualifications

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the areas of systems engineering, evaluation of the hazard types being considered for screening or conservative analysis, and evaluation of how the hazard types being considered for screening or conservative analysis could impact the nuclear plant's SSCs, as applicable to the scope of the review.

4.3.11.2.2 Review Technical Elements to Confirm the Methodology

4.3.11.2.2.1 Hazard Selection

The peer review shall focus on the potential for the hazard type to lead to release categories. The peer review team shall evaluate whether the hazard information is appropriately specific to the site and has met the relevant requirements of this Standard.

4.3.11.2.2.2 Hazard-Caused Initiating Events

The peer review team shall evaluate whether the plant initiating events postulated to be caused by the hazard type are identified, the SSCs considered in the screening assessment or conservative analysis are modeled, and any event sequences included are quantified.

4.3.11.2.2.3 Screening

The peer review team shall evaluate whether the basis for applying any qualitative, demonstrably conservative, and/or quantitative screening criteria is appropriately specific to the site and has met the relevant requirements of this Standard.

The peer review team shall evaluate whether the quantification method used in the quantitative screening analysis is appropriate and is capable of providing insights needed for risk-informed decisions. The peer review team shall review the validity of the screening assumptions.

4.3.11.2.2.4 Investigation

The peer review team shall review the investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable] of the plant and its surroundings, as applicable, to ensure the validity of the findings of the analysis in terms of screening, any spatial interactions, and the identification of critical failure modes.

4.3.11.3 References for Hazards Screening Analysis

The following is a list of publications referenced in this Standard.

[HS-1] NUREG/CR-2300, J. HICKMAN et al., "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," American Nuclear Society/Institute of Electrical and Electronic Engineers/U.S. Nuclear Regulatory Commission, 1983

[HS-2] NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities," U.S. Nuclear Regulatory Commission, 1991

[HS-3] NUREG/CR-4550, J. A. LAMBRIGHT et al., Vol. 4, Rev. 1, Part 3, "Analysis of Core Damage Frequency: Peach Bottom, Unit 2 External Events," Sandia National Laboratories/U.S. Nuclear Regulatory Commission, 1990

[HS-4] NUREG-75/087, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," U.S. Nuclear Regulatory Commission, 1975

[HS-5] Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, 2007

[HS-6] Regulatory Guide 1.91, "Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants," U.S. Nuclear Regulatory Commission, 2013

[HS-7] Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Reactors," U.S. Nuclear Regulatory Commission, 2007

[HS-8] Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, 1976 (errata in 1980)

[HS-9] NUREG/CR-4839, SAND87-7156, M. K. RAVINDRA and H. BANNON, "Methods for External Event Screening Quantification: Risk Methods Integration and Evaluation Program (RMIEP) Methods Development," Sandia National Laboratories/U.S. Nuclear Regulatory Commission, 1992

[HS-10] NUREG/CR-4832, M. K. RAVINDRA and H. BANNON, "Analysis of the LaSalle Unit 2 Nuclear Power Plant: Risk Methods Integration and Evaluation Program (RMIEP)," Vol. 7, "External Event Scoping Quantification," Sandia National Laboratories/U.S. Nuclear Regulatory Commission, 1992

[HS-11] NUREG/CR-5042, UCID-21223, C. Y. KIMURA and R. J. BUDNITZ, "Evaluation of External Hazards to Nuclear Power Plants in the United States," Lawrence Livermore National Laboratory/U.S. Nuclear Regulatory Commission, 1987

NONMANDATORY APPENDIX HS: NOTES AND EXPLANATORY MATERIAL FOR HAZARDS SCREENING ANALYSIS

HS.1 NOTES ASSOCIATED WITH HAZARDS SCREENING ANALYSIS

Table HS-1 Notes Supporting Hazards Screening Analysis Requirements

Number	Notes
HS-N-1	The description of a bounding site can be the identification of an existing site if that site bounds the hazard for other sites under consideration. See HS-A1
HS-N-2	Human-induced earthquakes (e.g., due to extraction of fossil fuels, mining activities) are screened as appropriate using Hazards Screening Analysis (natural tectonic earthquakes are addressed in seismic PRA). The following are sample criteria suggested for consideration: *(a)* The closest distance between the site and the location of recorded earthquakes that are considered as seismicity that is induced or triggered by human activities is greater than 200 miles from the site. *(b)* The magnitude of induced or triggered earthquakes is below that used to derive the earthquake recurrence models, implying that the suite of recurrence models used for the probabilistic seismic hazard analysis remain appropriate. *(c)* The rate of induced or triggered earthquakes would not increase the mean recurrence rate for any of the seismic sources that are within 200 miles of the site by more than 10%. *(d)* Median ground motions estimated using the closest distance and the maximum expected magnitude for induced or triggered events is less than ground motion at a mean annual frequency of exceedance of 10^{-3}. Ground motions should be evaluated for both peak ground acceleration and 10 Hz spectral acceleration. See HS-A2
HS-N-3	Design-unique hazards may include chemical hazards such as sodium-potassium eutectic fires from a decay heat removal loop. See HS-A3
HS-N-4	The purpose of this requirement is to ensure that an unusual type of hazard is not inadvertently omitted simply because it does not fit into any of the list of hazards commonly considered and listed in the standard references in Requirement HS-A2. See HS-A3, HS-A4
HS-N-5	HS-A3 focuses on identifying -plant-specific hazards that are not inherent to the design or the site and can only be identified during or after plant construction (e.g., a decision to co-locate a chemical plant on the same site as the reactor). See HS-A4
HS-N-6	This SR is likely not applicable to PRAs performed during the pre-operational stage. See HS-A4, HS-B2, HS-B3
HS-N-7	In the siting and plant pre-operational stage, most site-specific natural and man-made hazards will have been addressed and included in the design basis, unless they were screened out using the licensing criteria described in the NRC Standard Review Plan (SRP) and RGs. Such documented information can be useful input and reference information in the Hazards Screening Analysis screening process. See HS-B1, HS-B2
HS-N-8	Items *(a)* through *(f)* of the list in this SR are specifically identified because they represent the most common areas where a significant change might have occurred since the issuance of the operating license. See HS-B3

Table HS-1 Notes Supporting Hazards Screening Analysis Requirements (Cont'd)

Number	Notes
HS-N-9	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See HS-B7, HS-C14
HS-N-10	This SR is not applicable to operating plants. See HS-B7, HS-C14, HS-E3
HS-N-11	As the part of the development of the hazard screening plant response model, consideration should be given to whether new initiating events, unmodeled plant conditions, or event sequences may need to be added to represent the impacts of the hazard for the range of magnitudes under consideration. For example, multiple failures coupled with previously unmodeled plant conditions or plant response may result in unexpected outcomes for hazard events of different magnitudes. See HS-C4
HS-N-12	If a hazard is being screened that has a technical element in this Standard (i.e., internal fire, internal flood, high wind, external flood), a demonstrably conservative implementation of the initiating events, event sequences, and the systems logic models for that hazard may be leveraged as the basis for the hazard screening plant response model instead of relying solely on the initiating events model. All of these hazard plant response models use the initiating events model as the basis for that hazards plant response model. For all other hazards not directly addressed in this Standard, the event sequences and the systems logic model from the internal events PRA model should be used as the basis for the hazard screening plant response model. See HS-C4
HS-N-13	The term disposition means that peer review exceptions and deficiencies (e.g., Facts and Observations) means a given exception or deficiency has either been resolved (i.e., closed) or has been shown to not impact the plant response model. See HS-C5
HS-N-14	There is a single Capability Category for Hazards Screening Analysis. Thus, new system modeling would need to meet Capability Category I (CC-I) for any screening model. See HS-C6

Table HS-1 Notes Supporting Hazards Screening Analysis Requirements (Cont'd)

Number	Notes
HS-N-15	It is important to recognize that a demonstrably conservative estimate of a mean value is not a point estimate. When uncertainties are large, the mean frequency can fall above the 95th percentile of the distribution. Therefore, it is incumbent on the analyst to document the evidence that justifies estimates of uncertainties, approximations, or simplifications leading to the estimate of the mean event sequence frequencies. Calculation of the event sequence frequency may be done using different demonstrably conservative assumptions, as explained by the following example. Typically, nuclear power plants are sited such that the accidental impact of plant structures by aircraft is highly unlikely. As part of the hazard PRA, the risk from aircraft events may be assessed at different levels. The mean annual frequency of aircraft impact during takeoff, landing, or in flight may be determined. If this hazard frequency is very low, then the aircraft impact as a hazard may be eliminated from further study. This approach assumes that the aircraft impact results in damage of the structure leading to radionuclide release (this assumption is likely to be highly conservative). If the frequency of aircraft impacting the plant structures is estimated to be larger, the fragility of the structures may be evaluated to make a refined estimate of the frequency of radionuclide release. Further refinements could include the following: *(a)* eliminating certain structural failures as not resulting in radionuclide release (e.g., damage of diesel generator building may not result in radionuclide release if off-site electrical power is available); *(b)* performing a plant systems and Event Sequence Analysis to calculate the event sequence frequencies. This example shows that for some hazards, it may be sufficient to perform only the hazard analysis; for some others, the hazard analysis and a simple fragility analysis may be needed; in rare cases, a plant systems and Event Sequence Analysis may be necessary. Other examples of demonstrably conservative analysis can be found in references [HS-4], [HS-6], [HS-7], and [HS-8]. See HS-C7
HS-N-16	It is not expected that a new consequence analysis will need to be conducted to the support quantitative hazard screening process. Instead, the event sequences families resulting from the quantitative hazards screening process should be conservatively mapped to existing consequence estimates for existing event sequence families. See HS-C10
HS-N-17	The results of the screening model will be incorporated in the Risk-Integration technical element. See HS-C12
HS-N-18	Examples of investigations include, but are not limited to, actives such as tabletop review(s), physical/computerized walkdown(s) of the plant, and/or interview(s) with knowledgeable personnel. These investigation(s) may need to be supported by a review of plant documents, especially in the context of external hazard evaluations. This SR does not imply that each of these activities need to be performed to meet the SR. Further, if appropriate, one investigation can meet the intent of multiple investigation SRs if the investigation is scoped and structured appropriately. See HS-D1
HS-N-19	The general hazard screening investigations should concentrate, although not exclusively, on outdoor facilities that could be affected by on-site hazards (e.g., on-site storage of hazardous materials) and off-site developments such as increased usage of new airports/airways, highways, and gas pipelines. The purpose of this HS-D1 is to direct the analyst to look beyond the plant licensing documents. See HS-D1
HS-N-20	This SR is not applicable to PRAs performed for a specific site. See HS-E4

HS.2 Notes Associated With Hazards Screening Analysis High Level Requirements

The following table provides a typical list of internal and external hazard groups and their associated hazards. This list of hazards is compiled based on review of industry studies such as NUREG/CR-2300 [HS-1], NUREG-1407 [HS-3], IAEA SSG-3 [HS-5], NUREG/CR-5042 [HS-8], EPRI 1022997 [HS-9], EPRI 3002005287 [HS-10], and ASAMPSA_E List of External Hazards [HS-11]. Note that some studies identify hazards broadly (e.g., chemical release) whereas other studies more specifically (e.g., ground contamination from chemicals, chemical release into water), and some studies provide miscellaneous hazards not listed in the table below (e.g., corrosion, solar storm, air pollution and mist). This table does not explicitly list internal events, internal flooding, and internal fires.

Table HS-2 List of Hazards for Consideration

Hazard Group [Note (1)]	Hazard	Remarks [Notes (2), (3), (4)]
Animals	Animals	Land or flying animals can cause damage to plant equipment [such as loss of off-site power (LOOP)] or result in other hazards (such as transportation accidents). Impact on intake water from fish, mussels, and water-borne items are addressed by other hazards below.
Biological events	Biological events	Includes events such as detritus and zebra mussels blocking intake structure screens.
External fire	Forest fire	Plant design and fire-protection provisions often are adequate to mitigate the effects; however, site-specific analyses may be necessary to evaluate fire propagation (e.g., airborne fire brand transport).
	Grass fire	Fire often cannot propagate to or on the site because the site is cleared; plant design and fire-protection provisions are adequate to mitigate the effects; however, this can be confirmed via walkdown.
	Non-safety building fire	Fire often cannot propagate to safety areas of plant; separation, plant design, and fire-protection provisions are often adequate to mitigate the effects; however, this can be confirmed via walkdowns.
Extraterrestrial events	Meteorite or satellite impact	In the case of a low likelihood hazard; however, effects are not limited to direct impact but also include other related potential effects of indirect impacts or airburst events (e.g., total thermal exposure, overpressure, seismic event, ejecta, etc.).
Extreme temperature	Frost	Subsumed in snow and ice hazards.
	High summer temperature	Can often exclude where the ultimate heat sink is designed for at least 30 days of operation, including evaporation, drift, seepage, and other water-loss mechanisms. Evaluation is needed of possible loss of air cooling due to high temperatures.
	Ice cover	Ice blockage of river is included in flood; loss of cooling-water flow is considered in plant design.
	Low winter temperature	Thermal stresses and embrittlement are usually insignificant or covered by design codes and standards for plant design; generally, there is adequate warning of icing on the ultimate heat sink so that remedial action can be taken. However, the reliability of operator actions and equipment used to protect vulnerable SSCs (e.g., heat tracing on water-carrying pipes) may need to be evaluated.
Ground shifts	Avalanche (rock or debris)	Can be excluded for most nuclear plant sites; confirm through siting review or walkdown.
	Coastal erosion	Included in the effects of External Flooding PRA.
	Landslide	Can be excluded through siting review; confirm through walkdown.
	Sinkholes	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard.
	Soil shrink–swell	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard.

Table HS-2 List of Hazards for Consideration (Cont'd)

Hazard Group [Note (1)]	Hazard	Remarks [Notes (2), (3), (4)]
Heat-sink effects	Drought	Can often be excluded where there are multiple sources of ultimate heat sink or where the ultimate heat sink is not affected by drought (e.g., cooling tower with adequately sized basin).
	Frazil ice	Frazil ice is a slush of ice crystals that can rapidly form in turbulent water. Site-specific.
	Low lake or river water level	May result from failure of downstream dam. Can often be excluded where the ultimate heat sink is designed for at least 30 days of operation, including evaporation, drift, seepage, and other water-loss mechanisms if there is no downstream dam failure.
	River diversion	Considered in the evaluation of the ultimate heat sink; should diversion become a hazard, adequate storage is usually provided.
Heavy load drop	Heavy-load drop	Site-specific; requires detailed study.
High winds	Straight winds	Site-specific; requires detailed study.
	Tornadoes	Site-specific; requires detailed study.
	Tropical cyclones (i.e., hurricane, typhoon)	Involves both wind forces and external flooding. Wind forces covered under extreme winds and tornadoes.
	Sandstorm	Note that potential blockage of air intakes with particulate matter is generally considered in plant design; however, other adverse effects may need to be considered (e.g., particulate intrusion into electrical equipment).
	Hail	Other missiles govern.
Industrial accidents	Industrial or military facility accident	Includes externally generated missiles. Site-specific. May be screened based on proximity to site.
	Pipeline accident	May include both chemical release and/or explosion. May be screened based on proximity to site and content of pipeline. Site-specific.
	Release of chemicals from on-site storage	Plant-specific; requires detailed study.
	On-site excavation work	Temporary condition. Site-specific.
	Toxic gas	Site-specific; requires detailed study.
Lightning	Lightning	Considered in plant design; may not trip plant; LOOP often includes this contributor.
Seismic	Natural tectonic earthquakes	Site-specific; requires detailed study. See Seismic PRA.
	Human-induced earthquakes	Includes such causes as extraction of fossil fuels and mining activities.
External flooding	High tide	Included under external flooding.
	Precipitation, intense	Included under external and internal flooding. See External Flooding PRA for screening and detailed PRA.
	Seiche	Included under external flooding.
	Storm surge	Included under external flooding.
	Tsunami	Included under external flooding and seismic events. See Seismic PRA and External Flooding PRA for screening and detailed PRA.
	Waves	Included under external flooding.
Snow	Snow	Plant designed for higher loading. Regional climatology influences plant-specific susceptibility. Snowmelt causing river flooding is included under that flood hazard.
	Avalanche (snow)	Can be excluded for most nuclear plant sites; confirm through walkdown.
Transportation accidents	Aircraft impacts	Site-specific; requires detailed study.
	Fog	Could increase the frequency of man-made hazard involving surface vehicles or aircraft; accident data include the effects.
	Ship impact	Site-specific; requires detailed study.
	Vehicle impact	Plant-specific; requires detailed study.
	Railcar impact	Plant-specific; requires detailed study.
	Vehicle, railway car or ship explosion	Plant-specific; requires detailed study.

Table HS-2 List of Hazards for Consideration (Cont'd)

Hazard Group [Note (1)]	Hazard	Remarks [Notes (2), (3), (4)]
Site-generated missiles	Turbine-generated missiles	Plant-specific configuration issue.
	Other internally-generated missiles	Plant-specific configuration issue.
Volcanic activity	Volcanic activity	Can be excluded for most sites; however, distant impacts of an event may need to be considered (e.g., ash fallout, seismic events).

NOTES:

(1) The occurrence of any listed hazard that results from sabotage or terrorism is excluded from consideration.

(2) The statements in the remarks column have been typical of past approaches.

(3) The screening guidance provided here only addresses screening out of hazards using the criteria in Requirement HS-B1 (and Requirement HS-B2, if applicable). The remark "Site-specific; requires detailed study" should not be taken to imply that PRA using the requirements is required. Rather, detailed study could be limited to demonstrating that the hazard can be screened out using the criteria in Requirement HS-C1.

(4) The idea behind the screening remark that a given hazard is screened because it is "included under" or "covered by" another hazard is that it is not evaluated separately but is inherently included in another data set.

4.3.12 High Winds PRA (W)

This Section presents the technical requirements associated with High Winds PRA.

The requirements in this Section are divided into the following technical subelements:

(a) Wind Hazard Analysis (WHA);
(b) Wind Fragility Analysis (WFR);
(c) Wind Plant Response Analysis (WPR).

4.3.12.1 Objectives and Technical Requirements for Wind Hazard Analysis (WHA)

The objectives of the Wind Hazard Analysis ensure that

(a) high wind screening is performed;
(b) wind speed frequencies are determined;
(c) straight wind speeds represent the region or site;
(d) tropical cyclone represent the region or site;
(e) tornadoes represent the region or site;
(f) propagation of aleatory and epistemic uncertainties in each step of the probabilistic wind hazard analysis is performed; and
(g) the Wind Hazard Analysis is documented so as to provide traceability of the work.

Table 4.3.12.1-1 High Level Requirements for Wind Hazard Analysis

Designator	Requirement
HLR-WHA-A	Screening out of tropical cyclones, straight winds, and tornadoes, as applicable to the site, shall use wind data, site, and plant characteristics.
HLR-WHA-B	The frequencies of wind speeds at the sites shall be based upon probabilistic wind hazard analysis.
HLR-WHA-C	The probabilistic wind hazard analysis for straight wind speeds (e.g., thunderstorms and extra-tropical cyclones) at the site shall represent applicable regional and site-specific information.
HLR-WHA-D	The probabilistic wind hazard analysis for tropical cyclones shall represent applicable regional and site-specific information.
HLR-WHA-E	The probabilistic wind hazard analysis for tornadoes shall represent applicable regional and site-specific information.
HLR-WHA-F	Aleatory and epistemic uncertainties in the probabilistic wind hazard analysis shall be identified, characterized, propagated, and included in the final quantification of hazard estimates for the site.
HLR-WHA-G	The documentation of the Wind Hazard Analysis shall provide traceability of the work.

Table 4.3.12.1-2 Supporting Requirements for HLR-WHA-A

Screening out of tropical cyclones, straight winds, and tornadoes, as applicable to the site, shall use wind data, site, and plant characteristics. (HLR-WHA-A)

Index No. WHA-A	Capability Category I	Capability Category II
WHA-A1	For the Wind Hazard Analysis, either: *(a)* IDENTIFY the site at which the reactor being analyzed is located; or *(b)* DESCRIBE a bounding site and JUSTIFY that the bounding site bounds the list of sites in the scope of the PRA. See Note W-N-1	
WHA-A2	COMPILE a list of high wind hazards, including combinations of wind hazards that are applicable to the site. See Note W-N-2, W-N-3, W-N-4	
WHA-A3	COLLECT current data and information for the site and region to be used for the high wind hazard screening. See Note W-N-5	
WHA-A4	ENSURE that the site meets the probabilistic screening criteria SCR-1 or SCR-2 in Table 1.10-1 when screening out straight winds from the probabilistic wind hazard analysis. This can use a demonstrably conservative assessment or a realistic assessment that meets all requirements of Wind Hazard Analysis, Wind Fragility Analysis, and Wind Plant Response Analysis. See Note W-N-6, W-N-7	

Table 4.3.12.1-2 Supporting Requirements for HLR-WHA-A (Cont'd)

Screening out of tropical cyclones, straight winds, and tornadoes, as applicable to the site, shall use wind data, site, and plant characteristics. (HLR-WHA-A)

Index No. WHA-A	Capability Category I	Capability Category II
WHA-A5	ENSURE that the site meets one of the following conditions when screening out tropical cyclone (hurricane or typhoon) high wind hazards from the probabilistic wind hazard analysis: *(a)* meet SCR-3 in Table 1.10-1 by showing that the site is more than 150 miles (approximately 250 km) from the nearest tropical cyclone-prone coast to screen out tropical cyclone (hurricane or typhoon) high wind hazards from the probabilistic wind hazard analysis; *(b)* using either a demonstrably conservative assessment or a realistic assessment that meets all requirements of Wind Hazard Analysis, Wind Fragility Analysis, and Wind Plant Response Analysis, meet the hazard screening criteria of SCR-1 or SCR-2 in Table 1.10-1. See Note W-N-6, W-N-8	
WHA-A6	ENSURE that the site meets one of the following conditions when screening out tornado high wind hazards from the probabilistic wind hazard analysis: *(a)* meet SCR-3 in Table 1.10-1 by showing that, for a broad region surrounding the site, tornadoes have not occurred and the meteorological conditions for tornado genesis do not exist; *(b)* using either a demonstrably conservative assessment or a realistic assessment that meets all requirements of Wind Hazard Analysis, Wind Fragility Analysis, and Wind Plant Response Analysis, meet the hazard screening criteria of SCR-1 or SCR-2 in Table 1.10-1. See Note W-N-6, W-N-9	
WHA-A7	Using either a demonstrably conservative assessment or a realistic assessment that meets all applicable requirements of Wind Hazard Analysis, Wind Fragility Analysis, and Wind Plant Response Analysis, ENSURE the aggregate event sequence family frequencies of all high wind hazards probabilistically screened out from the high wind hazard group does not exceed the screening criteria SCR-2 or JUSTIFY the use of alternate criteria. See Note W-N-10	
WHA-A8	CONFIRM that the high wind hazard screening correctly represents for either the as-built, as-operated or as-designed, as-intended-to-operate plant conditions via investigation(s). SATISFY the Supporting Requirements (SRs) of HLR-HS-C and HLR-HS-D when relying on quantitative screening described in WHA-A4, WHA-A5, or WHA-A6. See Note W-N-11, W-N-12	

Table 4.3.12.1-3 Supporting Requirements for HLR-WHA-B

The frequencies of wind speeds at the sites shall be based upon probabilistic wind hazard analysis. (HLR-WHA-B)

Index No. WHA-B	Capability Category I	Capability Category II
WHA-B1	DEFINE the reference wind speed parameters for each high wind hazard and justify any deviations from applicable national wind loading standard. See Note W-N-13	
WHA-B2	When calculating reference wind speeds from raw wind speed data, APPLY currently accepted wind speed conversion methods. See Note W-N-14	
WHA-B3	ENSURE that the Wind Hazard Analysis includes the effect of short duration wind events, such as thunderstorms and tornadoes, in the derivation of the resulting wind speed frequencies. See Note W-N-15	
WHA-B4	SPECIFY a lower bound wind speed magnitude (V_L) for the probabilistic wind hazard analysis that provides assurance that potential high wind damage to structures, systems, and components (SSCs) identified on the high wind equipment list (HWEL) is captured. See Note W-N-16	
WHA-B5	ENSURE that the discretization of the high wind speed hazard curves into intervals produces sufficient information for accurate wind frequency and plant response determination over the full range of wind speeds $\geq V_L$. See Note W-N-17	
WHA-B6	In developing the high wind hazard results for use in Event Sequence Quantification, EXTEND the high wind speed to large enough values (consistent with the physical data and interpretations) so that the truncation does not affect final numerical results (e.g., on parameters such as event sequence family frequencies) and the delineation and ranking of high wind-induced sequences are not affected. See Note W-N-18	

Table 4.3.12.1-4 Supporting Requirements for HLR-WHA-C

The probabilistic wind hazard analysis for straight wind speeds (for example, thunderstorms and extra-tropical cyclones) at the site shall represent applicable regional and site-specific information. (HLR-WHA-C)

Index No. WHA-C	Capability Category I	Capability Category II
WHA-C1	IDENTIFY anemometer stations near the site and EVALUATE the applicability and quality of the wind data at each station for use in the probabilistic wind hazard analysis. See Note W-N-19, W-N-20	
WHA-C2	In analyzing wind station data, ENSURE that the data are updated, as necessary, to the reference wind speed defined in Requirement WHA-B1. See Note W-N-21	
WHA-C3	ANALYZE straight wind data without separation of thunderstorm from non-thunderstorm data. See Note W-N-22	ANALYZE thunderstorm and non-thunderstorm data separately and GROUP these separate frequencies to produce the straight wind hazard frequencies. See Note W-N-23
WHA-C4	JUSTIFY the distribution used for the wind speed probability in the analysis and its use in the context of rare straight wind phenomena. See Note W-N-24	
WHA-C5	JUSTIFY the method used to produce the site-specific straight wind frequencies from wind data records analyzed from one or more locations. See Note W-N-25	
WHA-C6	COMPARE the straight wind speed frequencies and uncertainties with reference to most recent published data, if available. IDENTIFY causes for significant differences. See Note W-N-26	

Table 4.3.12.1-5 Supporting Requirements for HLR-WHA-D

The probabilistic wind hazard analysis for tropical cyclones shall represent applicable regional and site-specific information. (HLR-WHA-D)

Index No. WHA-D	Capability Category I	Capability Category II
WHA-D1	DEVELOP site-specific probabilistic wind hazard analysis tropical cyclone wind speed frequencies using one of the following methods: *(a)* an analysis using data from a study, publication, or standard that meets SRs of HLR-WHA-G; or *(b)* calculations with probabilistic hurricane models that include: *(1)* frequency and intensity data; *(2)* spatial modeling of storm tracks; *(3)* validated wind field model; *(4)* validated wind pressure relationship; and *(5)* validated inland decay model. See Note W-N-27	
WHA-D2	COMPARE the tropical cyclone wind speed frequencies and uncertainties with reference to most recent published data, if available. IDENTIFY causes for significant differences. See Note W-N-28	

Table 4.3.12.1-6 Supporting Requirements for HLR-WHA-E

The probabilistic wind hazard analysis for tornadoes shall represent applicable regional and site-specific information. (HLR-WHA-E)

Index No. WHA-E	Capability Category I	Capability Category II
WHA-E1	DEVELOP site-specific probabilistic wind hazard analysis tornado wind speed frequencies using one of the following methods: *(a)* an analysis using data from a study, publication, or standard that meets the SRs of HLR-WHA-G and Requirement WHA-E2; or *(b)* calculations using a probabilistic tornado hazard model that meets Requirement WHA-E2. See Note W-N-29	
WHA-E2	ENSURE that the probabilistic wind hazard analysis tornado wind speed frequencies include *(a)* frequency analysis and intensity data that represents the site and regional tornado climatological risk; *(b)* analysis of and corrections for tornado reporting limitations and uncertainties; *(c)* tornado path length and width correlations to tornado intensity; *(d)* variation of tornado intensity along the path length and across the path width; and *(e)* probabilistic models of tornado wind speed given damage intensity rating. See Note W-N-30	
WHA-E3	ENSURE that the tornado region used in the site analysis is reasonably homogeneous and sufficiently broad to adequately represents the tornado climatology at the site and the risks associated with rare events.	
WHA-E4	ENSURE that tornado wind speed frequencies account for the effects of wind pressure, atmospheric pressure change, and wind-borne missiles. See Note W-N-31	
WHA-E5	COMPARE the tornado wind speed frequencies and uncertainties with reference to most recent published data, if available. IDENTIFY causes for significant differences. See Note W-N-32	

Table 4.3.12.1-7 Supporting Requirements for HLR-WHA-F

Aleatory and epistemic uncertainties in the probabilistic wind hazard analysis shall be identified, characterized, propagated, and included in the final quantification of hazard estimates for the site. (HLR-WHA-F)

Index No. WHA-F	Capability Category I	Capability Category II
WHA-F1	IDENTIFY assumptions and sources of uncertainties in each step of the Wind Hazard Analysis in a manner that supports Requirements WHA-F2 and WPR-E7. See Note W-N-33	
WHA-F2	CHARACTERIZE notable sources of uncertainty for each high wind hazard (e.g., using sensitivity studies related to alternate data, models, and methods).	
WHA-F3	ESTIMATE uncertainties in final quantification of the high wind hazard using demonstrably conservative assumptions. ASSESS how conservative assumptions affect key insights and conclusions.	PROPAGATE the uncertainties that are contributors to the high wind frequency quantifications. See Note W-N-34
WHA-F4	ESTIMATE representative hazard frequency functions for each type of high wind included in the model. JUSTIFY that the approach used yields a demonstrably conservative estimate of hazard frequencies and addresses notable sources of uncertainties. See WHA-F2	QUANTIFY hazard frequency functions, including mean and percentile hazard functions, for each type of high wind included in the model. See Note W-N-34

Table 4.3.12.1-8 Supporting Requirements for HLR-WHA-G

The documentation of the Wind Hazard Analysis shall provide traceability of the work. (HLR-WHA-G)

Index No. WHA-G	Capability Category I	Capability Category II
WHA-G1	DOCUMENT the process used in the Wind Hazard Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the process used to identify and screen out high wind hazard types; *(b)* the approach used to perform the probabilistic wind hazard analysis; *(c)* the data, models, and methods used for determining the high wind hazard curves; *(d)* the basis for including or excluding data, models, and methods in the analysis; *(e)* all assumptions.	
WHA-G2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives associated (as identified in Requirement WHA-F1) with the Wind Hazard Analysis.	
WHA-G3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the Wind Hazard Analysis (as identified in Requirement WHA-F1). See Note W-N-35	
WHA-G4	For Wind Hazard Analysis PRAs using a bounding site, DOCUMENT the basis for the selection of the bounding site characteristics that bound the range of sites. See Note W-N-36	

4.3.12.2 Objectives and Technical Requirements for Wind Fragility Analysis (WFR)

The objectives of the Wind Fragility Analysis ensure that

(a) wind fragilities of SSCs are incorporated for each wind hazard type whose failure can contribute to radiological release;
(b) data and findings from investigation(s) are incorporated to establish or confirm site conditions;
(c) screening is conducted of individual SSCs, wind effects, or failure modes;
(d) wind pressure and atmospheric pressure change (APC) effects are included;
(e) wind-generated missile effects are included;
(f) structural interactions effects are included;
(g) wind-driven rain effects, if relevant, are included;
(h) uncertainties are identified, propagated, and displayed; and
(i) the Wind Fragility Analysis is documented so as to provide traceability of the work.

Table 4.3.12.2-1 High Level Requirements for Wind Fragility Analysis

Designator	Requirement
HLR-WFR-A	The Wind Fragility Analysis shall incorporate wind fragilities of SSCs for each wind hazard type whose failure may contribute to event sequences or event sequence families.
HLR-WFR-B	The Wind Fragility Analysis shall incorporate the data and findings of investigation(s) to establish or confirm either as-built, as-operated or as-designed, as-intended-to-operate site conditions.
HLR-WFR-C	Fragility screening shall be based upon a structured process for individual SSCs, wind effects, and failure modes.
HLR-WFR-D	The Wind Fragility Analysis shall include wind pressure and APC effects.
HLR-WFR-E	The Wind Fragility Analysis shall include wind-generated missile effects.
HLR-WFR-F	The Wind Fragility Analysis shall include structural interactions effects.
HLR-WFR-G	The Wind Fragility Analysis shall include wind-driven rain effects if relevant to the plant.
HLR-WFR-H	Aleatory and epistemic uncertainties in each step of the Wind Fragility Analysis shall be identified, propagated, and displayed in the quantification of wind fragilities.
HLR-WFR-I	Documentation of the Wind Fragility Analysis shall provide traceability of the work.

Table 4.3.12.2-2 Supporting Requirements for HLR-WFR-A

The Wind Fragility Analysis shall incorporate wind fragilities of SSCs for each wind hazard type whose failure may contribute to event sequences or event sequence families. (HLR-WFR-A)

Index No. WFR-A	Capability Category I	Capability Category II
WFR-A1	INCLUDE in the scope of the Wind Fragility Analysis those SSCs and associated failure modes identified in the plant response analysis. See Requirements WPR-C1, WPR-C2, WPR-C3, and WPR-C4 See Note W-N-36, W-N-37	
WFR-A2	DEVELOP wind fragilities that are *(a)* based on the reference wind speed for each high wind hazard; *(b)* SSC-specific.	
WFR-A3	For PRAs conducted on a specific site, ENSURE that the wind fragilities are site-specific. See Note W-N-37	
WFR-A4	ENSURE that the wind fragilities are for the range of wind speeds developed in Requirements WHA-B5 and WHA-B6. See Note W-N-35, W-N-38	
WFR-A5	ENSURE that the significant failure modes for SSCs are included for each wind loading effect. See Note W-N-37, W-N-39	
WFR-A6	When multiple effects and/or failure modes are aggregated into a single fragility, JUSTIFY the method used for the aggregation. See Note W-N-37, W-N-40	

Table 4.3.12.2-2 Supporting Requirements for HLR-WFR-A (Cont'd)

The Wind Fragility Analysis shall incorporate wind fragilities of SSCs for each wind hazard type whose failure may contribute to event sequences or event sequence families. (HLR-WFR-A)

Index No. WFR-A	Capability Category I	Capability Category II
WFR-A7	When the same wind fragilities are used for different wind hazards, JUSTIFY the basis for not using wind-hazard-specific fragilities. See Note W-N-37, W-N-41	
WFR-A8	ASSESS high wind fragility correlations for their impact on high winds PRA results and insights. See Note W-N-37, W-N-42	
WFR-A9	ADDRESS the effects of coexistent hazards on the fragilities that are included in the high winds PRA scope, if applicable. See Note W-N-37, W-N-43, W-N-44	

Table 4.3.12.2-3 Supporting Requirements for HLR-WFR-B

The Wind Fragility Analysis shall incorporate the data and findings of investigation(s) to establish or confirm either as-built, as-operated or as-designed, as-intended-to-operate site conditions. (HLR-WFR-B)

Index No. WFR-B	Capability Category I	Capability Category II
WFR-B1	COLLECT information about either the as-built, as-operated, or as-designed, as-intended-to-operate, as applicable, site characteristics relevant to the fragility analysis, such as construction characteristics and potential failure modes for each wind effect by conducting investigation(s). See Note W-N-12, W-N-45	
WFR-B2	ENSURE that, for those SSCs included in HWEL, SSC supporting elements (such as associated piping, conduits, vents, supports, and other components required to support functionality) are identified in the investigation(s) and are included in the Wind Fragility Analysis. See Note W-N-12, W-N-38	
WFR-B3	In evaluating SSCs that are screened out for one or more wind failure modes, CONFIRM that the assumptions used in the screening analysis are consistent with the observations from the investigation(s). See Note W-N-12, W-N-38, W-N-46	
WFR-B4	For PRAs conducted on a specific site, COMPILE the number, types, and locations of potential missiles that may impact and fail individual SSCs (e.g., via plant survey). See Note W-N-47	
WFR-B5	For PRAs conducted on a bounding site, ESTIMATE the number, types, and locations of potential missiles that may impact and fail individual SSCs (e.g., via plant survey, tabletop review, computerized walkdown). See Note W-N-36	
WFR-B6	ENSURE that the missile characterization is consistent with the missile fragility methodology requirements under Requirements WFR-E3, WFR-E4, and WFR-E5.	
WFR-B7	ESTIMATE the number of potential missiles and their locations for different plant operating states, such as outage and non-outage modes. See Note W-N-38, W-N-48	

Table 4.3.12.2-4 Supporting Requirements for HLR-WFR-C

Fragility screening shall be based upon a structured process for individual SSCs, wind effects, and failure modes. (HLR-WFR-C)

Index No. WFR-C	Capability Category I	Capability Category II
WFR-C1	If an SSC is screened out for wind pressure effects and/or atmospheric pressure change effects, JUSTIFY the basis for the screening out evaluation. See Note W-N-38, W-N-49	
WFR-C2	If wind-generated missile effects are screened out for an SSC, JUSTIFY the methodology used and the basis for the screening out evaluation. See Note W-N-38, W-N-50	
WFR-C3	If structural interaction effects are screened out for an SSC, JUSTIFY the methodology used and the basis for the screening out evaluation. See Note W-N-38	
WFR-C4	If wind-driven rain effects are screened out for an SSC, JUSTIFY the methodology used and the basis for the screening out evaluation. See Note W-N-38, W-N-51	

Table 4.3.12.2-5 Supporting Requirements for HLR-WFR-D

The Wind Fragility Analysis shall include wind pressure and APC effects. (HLR-WFR-D)

Index No. WFR-D	Capability Category I	Capability Category II
WFR-D1	JUSTIFY the methods used for developing wind pressure load effects if they deviate from applicable national wind loading standards. See Note W-N-52	
WFR-D2	JUSTIFY the methods used for developing APC load effects, including methods for combining wind pressure and APC loads. See Note W-N-53	
WFR-D3	ENSURE that differences in wind design loads are included when the SSC design information and applicable codes are compared to current wind standards and codes. See Note W-N-54	
WFR-D4	If the SSC is a flexible structure, ENSURE that the dynamic response characteristics are factored into the wind effects. See Note W-N-55	
WFR-D5	EVALUATE the site for potential topographic effects according to applicable national standards or other published methodologies and, if applicable, ENSURE that the wind pressure effects incorporate topographic speed-ups. See Note W-N-56	
WFR-D6	EVALUATE SSCs for potential wind pressure load shielding/negative shielding effects, and INCLUDE factors for these potential effects in the fragility calculation, if applicable. See Note W-N-57	

Table 4.3.12.2-6 Supporting Requirements for HLR-WFR-E

The Wind Fragility Analysis shall include wind-generated missile effects. (HLR-WFR-E)

Index No. WFR-E	Capability Category I	Capability Category II
WFR-E1	USE the site-specific wind hazard characteristics and their associated wind fields for developing wind-generated missile effects. See Note W-N-58	
WFR-E2	JUSTIFY the basis for using missile effects that are not wind hazard-specific. See Note W-N-59	
WFR-E3	CATEGORIZE the missile types for fragility analysis using the numbers and types of missiles surveyed in Requirements WFR-B4, WFR-B6, and WFR-B7. See Note W-N-60	
WFR-E4	DESCRIBE how missiles from structure sources, including building envelope sources, building contents, and roof-top missiles are quantified and included in the missile analysis. See Note W-N-61	
WFR-E5	When missile sources that are beyond a certain distance from the nearest SSC are excluded from the analysis, JUSTIFY the basis. See Note W-N-62	
WFR-E6	When the missile impact and damage methodology uses scaling approaches based upon SSC dimensions, area, or volume, JUSTIFY the approach. See Note W-N-63	
WFR-E7	DEMONSTRATE that the missile impact and damage methodology produces stable numerical results for the missile effects over the range of wind speeds. See Note W-N-64	
WFR-E8	SPECIFY the assumptions and analysis methods in the missile effects analysis methodology, including the following: *(a)* the spatial effects of the plant layout, topography, SSC locations, and missile numbers and sources; *(b)* wind field characteristics; *(c)* missile injection, aerodynamics, and trajectory analysis; *(d)* missile impact and damage to SSCs; and *(e)* multiple missile generation in a wind hazard event. See Note W-N-65, W-N-66	SPECIFY the assumptions and analysis methods in the missile effects analysis methodology, including the following: *(a)* the spatial effects of the plant layout, topography, SSC locations, and missile numbers and sources; *(b)* shielding structures and features; *(c)* wind field characteristics; *(d)* missile injection, aerodynamics, and trajectory analysis, including ricochet into SSCs, if appropriate; *(e)* missile impact and damage to SSCs; and *(f)* multiple missile generation in a wind hazard event. When the spatial analysis components are not three dimensional, ENSURE that the method captures risk-significant and site-specific features. See Note W-N-67, W-N-68
WFR-E9	SATISFY the Capability Category I (CC-I) requirements of WFR-E8. See Note W-N-65, W-N-66	For PRAs conducted on a specific site, SATISFY the Capability Category II (CC-II) requirements of WFR-E8. ENSURE that the site-specific missile impact and damage calculations include the following: *(a)* site-specific wind hazard path sizes and path direction distributions; *(b)* missile type-dependent aerodynamics; and *(c)* missile damage analysis methods that depend on missile type. See Note W-N-47, W-N-67

Table 4.3.12.2-6 Supporting Requirements for HLR-WFR-E (Cont'd)

The Wind Fragility Analysis shall include wind-generated missile effects. (HLR-WFR-E)

Index No. WFR-E	Capability Category I	Capability Category II
WFR-E10	SPECIFY the missile hit/damage criterion for each SSC. See Note W-N-69	
WFR-E11	SPECIFY how the correlations of missile hit/damage to multiple SSCs in the same wind event are analyzed. See Note W-N-70	
WFR-E12	ENSURE that variations in missile populations (including outage/non-outage conditions and plant configuration changes) are included in the missile impact and damage analysis. See Note W-N-71	

Table 4.3.12.2-7 Supporting Requirements for HLR-WFR-F

The Wind Fragility Analysis shall include structural interactions effects. (HLR-WFR-F)

Index No. WFR-F	Capability Category I	Capability Category II
WFR-F1	DEFINE the methodology used for structural interaction analyses. See Note W-N-38, W-N-72	
WFR-F2	INCLUDE potential structural interaction effects from the failure of chimneys, stacks, exhausts, towers, poles, walls, roof structures, and other structures and components on SSCs included in the HWEL.	

Table 4.3.12.2-8 Supporting Requirements for HLR-WFR-G

The Wind Fragility Analysis shall include wind-driven rain effects if relevant to the plant. (HLR-WFR-G)

Index No. WFR-G	Capability Category I	Capability Category II
WFR-G1	DEFINE the methodology used for wind-driven rain effects. See Note W-N-73	
WFR-G2	INCLUDE the wind-driven rainwater entry paths that may lead to water drip, splash, and/or rain onto vulnerable SSCs. See Note W-N-74	

Table 4.3.12.2-9 Supporting Requirements for HLR-WFR-H

Aleatory and epistemic uncertainties in each step of the Wind Fragility Analysis shall be identified, propagated, and displayed in the quantification of wind fragilities. (HLR-WFR-H)

Index No. WFR-H	Capability Category I	Capability Category II
WFR-H1	ESTIMATE representative fragilities for failure modes included in the high wind plant response model. JUSTIFY that the fragilities are conservative and appropriate given the characteristics of the wind and SSC failure mechanisms. See Note W-N-75	CALCULATE fragilities for failure modes included in the high wind plant response model. JUSTIFY that the chosen form of each fragility is appropriate given the characteristics of the high wind and SSC failure mechanisms. See Note W-N-75
WFR-H2	In the development of SSC fragility, USE applicable generic information and technical evaluation. JUSTIFY that the use of generic information or technical evaluation is consistent with the conservative nature of the Category I assessment.	In the development of fragilities, USE *(a)* plant-specific information, if available; or *(b)* generic information augmented by a plant-specific technical evaluation. JUSTIFY that the use of generic information yields fragilities that are appropriate for the site.
WFR-H3	IDENTIFY assumptions and sources of uncertainty in the fragility evaluation in a manner that supports Requirements WFR-H4 and WPR-E7. See Note W-N-76	
WFR-H4	CHARACTERIZE notable sources of model uncertainty in the fragility evaluation. See Note W-N-76, W-N-77	

Table 4.3.12.2-10 Supporting Requirements for HLR-WFR-I

Documentation of the Wind Fragility Analysis shall provide traceability of the work. (HLR-WFR-I)

Index No. WFR-I	Capability Category I	Capability Category II
WFR-I1	DOCUMENT the process used in the Wind Fragility Analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the methodologies used to quantify the high wind fragilities of SSCs, along with assumptions; *(b)* a detailed set of SSC fragility values or fragility curves that includes the method of analysis, the dominant failure mode(s), the sources of information, and the location of each SSC; *(c)* the results of investigation(s); *(d)* the basis for screening out of any generic high-capacity SSCs; *(e)* parameter and fragility uncertainties.	
WFR-I2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement WFR-H3) associated with the Wind Fragility Analysis.	
WFR-I3	DOCUMENT assumptions and limitations due to the lack of as-built, as-operated design or site details for PRAs performed during pre-operational stage or pre-operational stage associated with the Wind Fragility Analysis (as identified in Requirement WFR-H3). See Note W-N-35	

4.3.12.3 Objectives and Technical Requirements for Wind Plant Response Analysis (WPR)

The objectives of the Wind Plant Response Analysis are to do the following:

(a) include high wind-caused initiating events in the high wind plant response model;

(b) include high wind-induced SSC failures, non-high wind induced SSC failures, unavailabilities, human errors, and multi-reactor effects that may lead to radioactive material release;

(c) include SSCs whose failure impact the high wind plant response model;

(d) incorporate high-wind specific challenges to credited human actions;

(e) quantify event sequences by integrating high wind hazard, fragilities, and plant response, including uncertainties; and

(f) document the Wind Plant Response Analysis so as to provide traceability of the work.

Table 4.3.12.3-1 High Level Requirements for Wind Plant Response Analysis

Designator	Requirement
HLR-WPR-A	The Wind Plant Response Analysis shall identify high wind induced initiating events that may result in risk-significant event sequences and/or risk-significant event progression sequences.
HLR-WPR-B	The Wind Plant Response Analysis shall include high wind-induced SSC failures, non-high wind induced SSC failures, unavailabilities, human errors, and multi-reactor effects that may lead to event sequence families.
HLR-WPR-C	The list of SSCs selected for Wind Plant Response Analysis shall include the SSCs that contribute to event sequences included in the high wind plant response model.
HLR-WPR-D	Human actions credited in the Wind Plant Response Analysis shall consider high wind-specific challenges to human performance.
HLR-WPR-E	The analysis to quantify event sequence family frequencies shall integrate the high wind hazard, the high wind fragilities, and the high wind plant response, including uncertainties.
HLR-WPR-F	Documentation of the Wind Plant Response Analysis shall provide traceability of the work.

Table 4.3.12.3-2 Supporting Requirements for HLR-WPR-A

The Wind Plant Response Analysis shall identify high wind induced initiating events that may result in risk-significant event sequences and/or risk-significant event progression sequences. (HLR-WPR-A)

Index No. WPR-A	Capability Category I	Capability Category II
WPR-A1	IDENTIFY high wind-induced initiating events caused directly by the high wind event, using a process that addresses the unique aspects of each applicable hazard type (e.g., straight wind, tornado, and tropical cyclone). See Note W-N-78	
WPR-A2	IDENTIFY initiating events caused directly or indirectly by the high wind event, including initiating events associated with changes in the plant mode (e.g., plant shutdown) or proceduralized plant reconfigurations (if applicable) due to the high wind event.	
WPR-A3	ENSURE the initiating events included in the Wind Plant Response Analysis incorporates industry experience. See Note W-N-79, W-N-80	
WPR-A4	INCLUDE in the plant response model the initiating events (e.g., due to failures of SSCs or human actions), identified in Requirements WPR-A1, WPR-A2, and WPR-A3, that cause risk-significant event sequences and/or risk-significant event progression sequences.	

Table 4.3.12.3-3 Supporting Requirements for HLR-WPR-B

The Wind Plant Response Analysis shall include high wind-induced SSC failures, non-high wind induced SSC failures, unavailabilities, human errors, and multi-reactor effects that may lead to event sequence families. (HLR-WPR-B)

Index No. WPR-B	Capability Category I	Capability Category II
WPR-B1	USE the event sequences and the systems logic model from the internal event PRA models as the basis of the high wind plant response model. INCLUDE additional event sequences, as applicable, associated with wind induced initiating events that are not included in the internal events PRA model such as events that impact two or more reactors or sources of radioactive material. See IE-A16 See Note W-N-81	
WPR-B2	ENSURE that the peer review findings for the internal events and other hazard PRAs that are relevant to the results of the high winds PRA are resolved and incorporated into the development of the Wind Plant Response Analysis model. See Note W-N-82	
WPR-B3	INCLUDE high wind-induced failures representing failure modes of interest in the Wind Plant Response Analysis. See Note W-N-83	
WPR-B4	MODEL the fragility correlation of wind-induced SSC failures, if applicable. JUSTIFY the use of correlation approach used. See Note W-N-84	
WPR-B5	JUSTIFY the inclusion of beneficial high wind induced SSC failures (e.g., by demonstrating exclusion of the failure would distort risk results). See Note W-N-85	

Table 4.3.12.3-3 Supporting Requirements for HLR-WPR-B (Cont'd)

The Wind Plant Response Analysis shall include high wind-induced SSC failures, non-high wind induced SSC failures, unavailabilities, human errors, and multi-reactor effects that may lead to event sequence families. (HLR-WPR-B)

Index No. WPR-B	Capability Category I	Capability Category II
WPR-B6	SPECIFY bounding scenario mission times consistent with establishing a stable end state of wind-induced risk-significant event sequences. See Note W-N-86	SPECIFY realistic scenario mission times consistent with relevant industry experience and the objective of establishing a stable end state of the wind-induced event sequences. For those events where the high winds mission time exceeds the equivalent internal events mission time, EXTEND the event sequence mission time to reach a stable end state. See Note W-N-86
WPR-B7	For new PRA logic models developed for the high winds PRA, SATISFY the following requirements, consistent with CC-I requirements (if applicable): *(a)* Initiating Event Analysis per HLR-IE-A and HLR-IE-B; *(b)* Event Sequence Analysis per HLR-ES-A and HLR-ES-B; *(c)* Success Criteria Development per HLR-SC-A and HLR-SC-B; *(d)* Systems Analysis per HLR-SY-A and HLR-SY-B; *(e)* Data Analysis per HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D; *(f)* use of expert judgment (if used; see Section 4.2). ENSURE the following are represented: *(a)* high wind-induced SSC failures; *(b)* SSC unavailabilities and failures not induced by the high wind event; and *(c)* human actions associated with high wind response (including high wind related actions not included within the internal events model) that can give rise to risk-significant event sequences or risk-significant event progression sequences.	For new PRA logic models developed for the high winds PRA, SATISFY the following requirements, consistent with CC-II requirements (if applicable): *(a)* Initiating Event Analysis per HLR-IE-A and HLR-IE-B; *(b)* Event Sequence Analysis per HLR-ES-A and HLR-ES-B; *(c)* Success Criteria Development per HLR-SC-A and HLR-SC-B; *(d)* Systems Analysis per HLR-SY-A and HLR-SY-B; *(e)* Data Analysis per HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D; *(f)* use of expert judgment (if used; see Section 4.2). ENSURE the following are represented: *(a)* high wind-induced SSC failures; *(b)* SSC unavailabilities and failures not induced by the high wind event; and *(c)* human actions associated with high wind response (including high wind related actions not included within the internal-events model) that can give rise to risk-significant event sequences or risk-significant event progression sequences.
WPR-B8	INCLUDE coexistent hazards that are within the scope of the high winds PRA. See Note W-N-43	
WPR-B9	For plant sites with multiple-reactors, ASSESS the effects of high winds on other reactors as it affects the reactor under study (e.g., effects on resources and organizational response, shared SSCs, CCFs, and site accessibility). See Note W-N-87, W-N-88	

Table 4.3.12.3-4 Supporting Requirements for HLR-WPR-C

The list of SSCs selected for Wind Plant Response Analysis shall include the SSCs that contribute to event sequences included in the high wind plant response model. (HLR-WPR-C)

Index No. WPR-C	Capability Category I	Capability Category II
WPR-C1	USE the internal events systems model developed against Requirement SY-A1 as the basis for developing a HWEL to support the fragility analysis. INCLUDE any additional systems that may have been incorporated into the high wind event sequence model in response to WPR-B1. See Note W-N-81, W-N-89	
WPR-C2	INCLUDE in the HWEL additional SSCs that may not be explicitly modeled in the internal-events model (or that may have been screened out in the internal events model) that require evaluation in the high winds PRA.	
WPR-C3	EXTEND the HWEL based on the review of industry high wind PRA HWELs, if available. See Note W-N-90	
WPR-C4	INCLUDE in the HWEL structural interactions (including spatial interactions) due to SSCs that may not be present in the internal events model. Structural interactions include impact of building collapse and physical contact of adjacent structures. As these impacts are driven by the proximity of structures, potential spatial interactions should be evaluated thoroughly during the walkdowns.	
WPR-C5	For the SSCs identified in Requirements WPR-C1, WPR-C2, WPR-C3, and WPR-C4, IDENTIFY the failure mode(s) of interest for the high Wind Fragility Analysis to be performed. See Note W-N-91	

Table 4.3.12.3-5 Supporting Requirements for HLR-WPR-D

Human actions credited in the High Winds PRA shall consider high wind-specific challenges to human performance. (HLR-WPR-D)

Index No. WPR-D	Capability Category I	Capability Category II
WPR-D1	IDENTIFY the human failure events (HFEs) (including preparatory and recovery actions) from the selected internal events PRA that are relevant in the context of the High Winds PRA.	
WPR-D2	EVALUATE operator actions for performance shaping factors related to unique aspects of each high wind hazard (e.g., straight line wind, tornado, and tropical cyclones). See Note W-N-92	
WPR-D3	For human response actions relevant to the Wind Plant Response Analysis, SATISFY the CC-I SRs of HLR-HR-E, except where the requirements are not applicable.	For human response actions relevant to the Wind Plant Response Analysis, SATISFY the CC-II SRs of HLR-HR-E, except where the requirements are not applicable.
WPR-D4	For definition and specification of HFEs for human response actions, SATISFY CC-I SRs of HLR-HR-F, except where the requirements are not applicable.	For definition and specification of HFEs for human response actions, SATISFY CC-II SRs of HLR-HR-F, except where the requirements are not applicable.
WPR-D5	For operating plants, REVIEW procedures and sequences of events with plant operations or training personnel to confirm that the interpretation of the procedures relevant to actions credited in the high winds PRA is consistent with plant operational and training practices. See Note W-N-93, W-N-94	

Table 4.3.12.3-5 Supporting Requirements for HLR-WPR-D (Cont'd)

Human actions credited in the High Winds PRA shall consider high wind-specific challenges to human performance. (HLR-WPR-D)

Index No. WPR-D	Capability Category I	Capability Category II
WPR-D6	For PRAs performed during the pre-operational stage, TALK THROUGH (i.e., review in detail) with knowledgeable personnel for plant operations or training personnel the assumed procedures and sequence of events to confirm that interpretation is consistent with plant operational and training practices. See Note W-N-93, W-N-95	
WPR-D7	For treatment of additional, exclusive recovery actions relevant to the Wind Plant Response Analysis, SATISFY SRs of HLR-HR-H, except where the requirements are not applicable.	
WPR-D8	INCLUDE HFEs in the high winds PRA plant response model such that the HFEs represent the impact of human failures at the function, system, train, or component level, as appropriate.	
WPR-D9	EVALUATE whether credited system recoveries modeled in the internal events PRA remain valid as a result of a high wind event.	
WPR-D10	ADJUST the credited recovery models based on results of Requirement WPR-D9. SPECIFY the basis for recovery values, if used (e.g., based on review of procedures and assessment of conditions under which actions will be performed).	
WPR-D11	For developing human error probabilities (HEPs), SATISFY the requirements at CC-I SRs of HLR-HR-G, except where they are not applicable, considering relevant high wind-related effects on control room and ex-control room actions. When addressing influencing factors and the timing considerations covered in Requirements HR-G4, HR-G6, and HR-G8, INCLUDE the effect of high wind hazard on the control room and ex-control room human actions. See Note W-N-96, W-N-97	For developing HEPs, SATISFY the requirements at CC-II SRs of HLR-HR-G, except where they are not applicable, considering relevant high wind-related effects on control room and ex-control room actions. When addressing influencing factors and the timing considerations covered in Requirements HR-G4, HR-G6, and HR-G8, INCLUDE the effect of high wind hazard on the control room and ex-control room human actions. See Note W-N-96, W-N-97

Table 4.3.12.3-6 Supporting Requirements for HLR-WPR-E

The analysis to quantify event sequence family frequencies shall integrate the high wind hazard, the high wind fragilities, and the high wind plant response, including uncertainties. (HLR-WPR-E)

Index No. WPR-E	Capability Category I	Capability Category II
WPR-E1	In the quantification of event sequence family frequencies on a plant-year basis, INTEGRATE the high wind hazard, fragility, and systems analyses in the PRA model. See Note W-N-98	
WPR-E2	ADDRESS overestimation of risk due to rare event approximations (e.g., where fragilities approach 1.0). See Note W-N-99	
WPR-E3	ENSURE that the discretization of the wind speed hazard curves (or other numerical methods used to incorporate the hazard curve in the integration) is appropriate to demonstrate convergence of risk metrics (e.g., the size and number of bins used to discretize the hazard curve). See Note W-N-100	

Table 4.3.12.3-6 Supporting Requirements for HLR-WPR-E (Cont'd)

The analysis to quantify event sequence family frequencies shall integrate the high wind hazard, the high wind fragilities, and the high wind plant response, including uncertainties. (HLR-WPR-E)

Index No. WPR-E	Capability Category I	Capability Category II
WPR-E4	When quantifying high wind hazard event sequence family frequencies, SATISFY the following requirements, except where the requirements are not applicable: *(a)* ESQ-A4, ESQ-A6, ESQ-A7; *(b)* ESQ-B1, ESQ-B2, ESQ-B3, ESQ-B5, ESQ-B6, ESQ-B7, ESQ-B8, ESQ-B9, ESQ-B10; *(c)* ESQ-C1, ESQ-C2, ESQ-C3, ESQ-C4, ESQ-C5, ESQ-C6, ESQ-C7, ESQ-C8, ESQ-C9, ESQ-C10, ESQ-C11, ESQ-C12, ESQ-C13, ESQ-C14, ESQ-C15, ESQ-C16, ESQ-C17; *(d)* ESQ-D1, ESQ-D2, ESQ-D3, ESQ-D5, ESQ-D6, and ESQ-D7. See Note W-N-101	
WPR-E5	USE the hazard curves, wind fragilities, and a point estimate quantification of plant response model to generate point estimates of event sequence family frequencies.	QUANTIFY the mean and the uncertainties of the event sequence family frequency estimates by propagating the uncertainties associated with high wind hazard frequency, high wind fragility, and high wind plant response model events through the quantification process.
WPR-E6	IDENTIFY assumptions and sources of uncertainty in the Wind Plant Response Analysis in a manner that supports Requirement WPR-E7.	
WPR-E7	CHARACTERIZE notable sources of uncertainty in the Wind Plant Response Analysis (e.g., using uncertainty analysis or sensitivity studies) and SATISFY ESQ-E1 for each high wind technical subelement as identified in Requirements WHA-F1 for Wind Hazard Analysis, WFR-H3 for WFR, and WPR-E6 for Wind Plant Response Analysis).	

Table 4.3.12.3-7 Supporting Requirements for HLR-WPR-F

Documentation of the high wind PRM and quantification analysis shall provide traceability of the work. (HLR-WPR-F)

Index No. WPR-F	Capability Category I	Capability Category II
WPR-F1	DOCUMENT the process used in the high wind PRM and quantification analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the specific adaptations made in the internal events PRA model to produce the high winds PRA model, and their bases; *(b)* those wind-related influences that affect methods, processes, or assumptions used, as well as the identification and quantification of the HFEs/HEPs in accordance with the SRs of HLR-WPR-D; and *(c)* the major outputs of a high winds PRA, such as event sequence frequency, event sequence family frequency, event sequence contributions, initiating event contributions, uncertainty distributions on event sequence family frequencies, results of sensitivity studies, and risk-significant contributors consistent with Requirements ESQ-F2, ESQ-F3, and ESQ-F4 except where the requirements are not applicable. See Note W-N-102	
WPR-F2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement WPR-E6) associated with the high wind plant response model and quantification analysis.	
WPR-F3	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details. As identified in Requirement WPR-E6. See Note W-N-35	

4.3.12.4 Peer Review Requirements for High Winds PRA

4.3.12.4.1 Purpose

This Section provides requirements for peer review of the High Wind Analysis element of the PRA.

4.3.12.4.2 Peer Review Team Composition and Personnel Qualifications

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the subjects of systems engineering, evaluation of the high wind hazard, and evaluation of how the high wind hazard effects could damage nuclear plant or design SSCs, as applicable to the scope of the review.

The requirements for individual team members and for specific review team qualifications consists of a comprehensive set of high wind technical activities and methodologies. Since high winds PRAs typically include multiple wind hazards (straight winds, tornadoes, and/or tropical cyclones, in particular) and effects (wind pressure, APC, wind-generated missiles, and wind-driven rain), the peer review requirement for direct experience in the specific methodologies requires breadth of experience. Furthermore, each of these hazards and effects encompasses distinct and typically extensive technical literature, accepted methodologies, disparate databases, methodologies, codes, tools, and approaches. Hence, additional peer reviewers, and/or specialized consultants to the peer review team, may be necessary to meet these requirements.

4.3.12.4.3 Review of High Wind PRA Elements to Confirm the Methodology

4.3.12.4.3.1 High Wind Hazard Review

The peer review team shall ensure that the high wind hazard types used in the PRA are comprehensive, appropriate to the bounding site covering a range of sites, and meet the relevant requirements of this Standard. The review team shall evaluate the reasonableness of the high wind hazard type results by comparing with currently published hazard data.

4.3.12.4.3.2 Wind-Induced Initiating Events

The peer review team shall evaluate whether the initiating events postulated to result from high wind events are properly identified, the SSCs are properly modeled, and the event sequences are properly quantified.

4.3.12.4.3.3 Fragility Analysis Methods

The peer review team shall evaluate whether the methods and data used in the fragility analysis are adequate for the purpose intended and meet the requirements of this Standard. As part of this evaluation, the peer review team should evaluate the adequacy and completeness of the following specific areas:

(a) identification and inclusion of SSC supporting equipment (such as piping, conduit, vents, etc.) in the fragility analysis;

(b) inclusion of plant or design outage and non-outage missile sources, structure failure missile sources, and roof-top missile sources (including ballast) in the missile characterization;

(c) understanding and documentation of SSC design information in relation to needed inputs for fragility analysis;

(d) dependency of wind fragilities on wind hazard type or the justification for using available wind fragilities not explicitly resulting from the hazard but derived from a somewhat similar one;

(e) applicable wind loading effects for each SSC;

(f) identification and analysis of risk-significant failure modes for each wind effect and SSC;

(g) consideration of wind-driven rain for vulnerable electrical equipment inside non-seismic Category I structures.

4.3.12.4.3.4 Investigation

The peer review team shall review the investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable] of the plant to ensure the validity of the findings in terms of: completeness of HWEL, critical failure modes, wind-generated missile sources, wind-generated missile pathways through openings and cladding, structural interactions, and wind-driven rain vulnerabilities.

For PRAs on plants prior to operation, if plant walkdown(s) is not possible the peer review team should review the findings of the following information via interviews and reviews (e.g., tabletop reviews, computerized simulations) with engineering personnel to ensure the validity of the findings in terms of completeness of HWEL, critical failure modes, wind-generated missile sources, wind-generated missile pathways through openings and cladding, structural interactions, and wind-driven rain vulnerabilities.

4.3.12.4.3.5 Quantification Method

The peer review team shall evaluate whether the quantification method used in the High Winds PRA is appropriate and provides the results and insights needed for risk-informed decision-making. If the analysis appears to contain questionable screening assumptions, or assumptions that the analysis team deems to be demonstrably conservative, the peer review team shall review the validity of these assumptions. The review shall focus on the event sequence and event sequence family frequency estimates and uncertainty bounds and on the risk-significant contributors.

4.3.12.5 References for High Winds PRA

The following is a list of publications referenced in this Standard.

[W-1] ASCE 7-98, "Minimum Design Loads for Buildings and Other Structures," American Society of Civil Engineers/Structural Engineering Institute, Reston, VA, 1998

[W-2] ASCE/SEI 7-10, "Minimum Design Loads for Buildings and Other Structures," American Society of Civil Engineers/Structural Engineering Institute, Reston, VA, 2010

[W-3] ASCE/SEI 7-16, "Minimum Design Loads for Buildings and Other Structures," American Society of Civil Engineers/Structural Engineering Institute, Reston, VA, 2016

[W-4] NUREG/CR-2815, Vol. 2, "Probabilistic Safety Analysis Procedures Guide," McCann, M., J. Reed, C. Ruger, K. Shiu, T. Teichmann, A. Unione, and R. Youngblood, Brookhaven National Laboratory and U.S. Nuclear Regulatory Commission; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852, 1985

[W-5] NUREG/CR-4458, "Extreme Wind Analysis for the Point Beach Nuclear Power Plant," Appendix G in "Shutdown Decay Heat Removal Analysis of a Westinghouse 2-Loop PWR;" Reed, J.W. and W. L. Ferrell; Sandia National Laboratories and U.S. Nuclear Regulatory Commission; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852, 1987

[W-6] NUREG/CR-4461, Rev. 2, "Tornado Climatology of the Contiguous United States," Ramsdell, J.V. and J. P Rishel; Battelle Pacific Northwest Laboratories and U.S. Nuclear Regulatory Commission; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852, 2007

[W-7] NUREG/CR-6372, "Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on the Use of Experts," R. J. Budnitz, G. Apostolakis, D. M. Boore, L. S. Cluff, K. J. Coppersmith, C. A. Cornell, and P. A. Morris; Lawrence Livermore National Laboratory and U.S. Nuclear Regulatory Commission; Publisher: U.S. Nuclear Regulatory Commission (NRC), One White Flint North, 11555 Rockville Pike, Rockville, MD 20852, 1997

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NONMANDATORY APPENDIX W: NOTES AND EXPLANATORY MATERIAL FOR HIGH WINDS PRA

W.1 NOTES ASSOCIATED WITH HIGH WINDS PRA

Table W-1 Notes Supporting High Winds PRA Requirements

Number	Notes
W-N-1	The description of a bounding site can be the identification of an existing site if that site bounds the hazard for other sites under consideration. See WHA-A1
W-N-2	Examples of potentially relevant high wind hazards include, but are not necessarily limited to, the following: *(a)* historical, regional, and site-specific high wind data; *(b)* high wind characteristics. See WHA-A2
W-N-3	Another example of potentially relevant high wind hazards includes high wind vulnerabilities (including major changes to the plant). See WHA-A2
W-N-4	Examples include, but are not limited to, the following: straight winds, tropical cyclones, tornadoes, and wind-driven rain. See WHA-A2
W-N-5	Relevant wind hazard data and analyses can often be found in national standards (such as ASCE 7 [W-1], [W-2], and [W-3] in the U.S.); NUREGs [W-4], [W-5], [W-6], [W-7], [W-8], [W-9], and [W-10]; National Oceanic and Atmospheric Administration (NOAA) National Climatic Data Center (NCDC) [W-11] and [W-12]; and the literature. In addition, site information is available at many plants. See WHA-A3
W-N-6	Probabilistic screening focuses on use of bounding/conservative hazard frequencies for screening but also considers the reliability of site protection and mitigation. When employing SCR-2, a consequence assessment is also needed to support the frequency screening threshold. See WHA-A4, WHA-A5, WHA-A6
W-N-7	Straight winds occur everywhere on earth. Extreme values of straight winds can exceed 125 mph and downburst wind speeds of 150 mph have been documented. Hence, deterministic screening is not allowed for straight winds, but they can be screened out using Probabilistic Screening. See WHA-A4
W-N-8	Distance Screening: The use of a 150-mile (approximately 250 km) screening distance was developed based on a review of multiple plant locations in the southeast U.S., where recent high winds PRAs have been performed with hurricane winds included. At about this distance, the hurricane high wind frequency falls below 20% of the total high wind frequency for wind speed bins of interest. The contribution to an LWR's core damage frequency was found to be less than a few percent. Hence, at this distance and beyond, tropical cyclones are expected to have an insignificant contribution to the high wind risks. See WHA-A5
W-N-9	Tornadoes may be screened from the high winds PRA using exclusion screening or probabilistic screening. These concepts are similar to the approaches allowed for tropical cyclone screening. Since tornadoes are not always reported and many countries may not have an official record for tornadoes, the analyst should be aware that the lack of reporting of tornadoes does not always mean that the hazard does not exist. For example, in regions that have severe thunderstorms, the meteorological conditions exist for tornadoes. Exclusion screening should therefore consider tornado reporting in nearby regions/countries with a similar climatology as well as the climatology of the region and the potential for tornadic conditions to be present. See WHA-A6

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-10	WHA-A7 is intended to represent the importance of ensuring that the aggregate risk from all high wind hazards is not significant when considering the selected risk metrics (e.g., ensure there are not a large number of high wind hazards that "barely screen" individually such that their aggregate contribution may be important). See WHA-A7
W-N-11	An investigation is required for performing probabilistic screening under WHA-A4, WHA-A5, and WHA-A6. It is important to note that SSCs that are pertinent to high wind events may not be identified in the internal events PRA. For example, the investigations should consider barriers, doors, off-site power lines, tanks, and other equipment uniquely related to the plant's high wind response (for example, see: Sciaudone et al. 2015 [W-13]; Lovelace et al. 2017a [W-14]. An example approach can be seen in the EPRI high wind walkdown guidance document: (EPRI 3002008092) [W-15]. The objective of the screening investigation is to support the required demonstrably conservative analysis. WHA-A7 does not require that the screening investigation meet the investigation requirements of HLR-WFR-B. Hence, for optional wind hazard screening, the PRA team should decide whether or not they are going to do a complete investigation according to the SRs under HLR-WFR-B. For example, if a simplified investigation is sufficient to support screening, but the screening is not successful for all wind hazards, then the PRA team may need to conduct a supplemental walkdown to satisfy the HLR-WFR-B SRs. See WHA-A8
W-N-12	Examples of investigations include, but are not limited to, actives such as tabletop review(s), physical/computerized walkdown(s) of the plant, and/or interview(s) with knowledgeable personnel. These investigation(s) may need to be supported by a review of plant documents, especially in the context of external hazard evaluations. This SR does not imply that each of these activities need to be performed to meet the SR. Further, if appropriate, one investigation can meet the intent of multiple investigation SRs if the investigation is scoped and structured appropriately. See WHA-A8, WFR-B1, WFR-B2, WFR-B3

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-13	The analyst must define the reference wind parameters for each wind hazard in the high winds PRA. Use of the same reference wind parameters for all wind hazards affecting a site simplifies the PRA and the fragility analysis. The parameters required to define the reference wind include the following: (a) averaging time, (b) surface roughness, (c) height above ground, and (d) direction. For example, in ASCE-7 [W-1], [W-2], and [W-3], the reference wind is specified as a three-second gust in open terrain at a 10-meter height above ground. The Canadian Code (NRCC, 2010) [W-16] uses hourly wind speeds, while the Australian Code (AS/NZS, 2011) [W-17] uses a 0.2-second peak gust, and the British Code (BSI, 2010) [W-18], which references the European standard, is based on a 10-minute average. It is important that the analyst understand the parameters of this Standard reference wind speed in the country/region in which the site is located so that informed decisions are made regarding the PHWA reference wind parameters. The reference wind generally includes all possible wind directions. That is, the reference wind is not based on a particular wind direction. However, note that in several recent high winds PRAs, directional wind analysis was used to support wind-generated missile analysis for straight winds and for wind-driven rain analysis. In these studies, the analysis of the wind data was performed to produce wind speed frequencies by directional octant, and these directional frequencies were used in the analysis. The results in Banik et al. (2017) [W-19] show that considering wind direction for straight winds can be used to reduce fragility conservatisms in a detailed modeling approach of building fragilities. An important connection between Wind Hazard Analysis and Wind Fragility Analysis is the use of reference wind as the independent variable for both hazard and fragility analysis. This approach, consistent with modern wind fragility modeling methods (e.g., ASCE 7 , Vickery et al. 2006 [W-20]; Pinelli et al. 2004 [W-21]; and Konthesingha, 2015 [W-22]), requires consideration of the wind effects (e.g., loads) and associated structural resistances in the fragility analysis. Note that wind effects/wind load considerations such as wind field characteristics, wind speed variation with height, site surface roughness, wind directional variation, topographic speed-ups, shielding/negative shielding effects, gust effects, and missile effects are appropriately included in the scope of the Wind Fragility Analysis and not in the Wind Hazard Analysis. This approach is consistent with national standards and standard wind engineering practice. See WHA-B1
W-N-14	Wind data may not always correspond to the analyst-defined reference wind parameters. In this case, the analyst will need to convert the wind data to reference wind conditions as part of the Wind Hazard Analysis. It is important to understand the details of the historical wind records at a site or meteorology station. Standards have changed over time, and the averaging time, exposure, and height of the anemometer can introduce significant biases into the frequency analysis. The sample frequency and averaging times of the measured wind speed are obtained from the source providing the wind speed data. For example, if the wind speed measurements are not continuous in time (e.g., periodic or hourly), the data likely do not contain the true daily or annual maxima, and it is likely that the wind hazard frequencies developed from such data will underestimate the true wind hazard frequencies. Vickery and Twisdale (2014) [W-23] provide an example of this situation by comparing results for a site anemometer with only hourly data to nearby airport station data, which includes peak gust wind speed measurements (also see the discussion under WHA-C1). All measured wind speeds need to be adjusted to a common averaging time. Gust factor curves such as those given in ASCE 7 [W-1] , [W-2], and [W-3] can be used to account for small averaging time differences. A useful reference for performing the adjustments for height, averaging times, and the effects of upstream terrain is Masters et al. (2010) [W-24]. See WHA-B2

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-15	For sites that are in regions where thunderstorms occur, the use of averaging times for straight wind that do not correspond to peak gusts can introduce significant underestimation errors in the Wind Hazard Analysis. In thunderstorm regions, if the available wind data do not include gust wind speeds obtained from a continuous record data sources that contain wind gust speed, data from areas with similar climatology may need to be considered. The reason for considering other data sources is that averaging times longer than a few minutes will filter out the effect of thunderstorms and other short duration wind events. The use of data from similar regions that do record peak gust information would likely enable the development of appropriate gust factors for the site. For sites located in regions where thunderstorms are not a significant part of the high winds PRA climatology, the use of longer averaging times than peak gust is acceptable for the straight wind analysis. In this case, peak gust factors derived from extratropical cyclones can be used to convert to peak gust winds, as needed. Tornado wind speeds are generally assumed to correspond to peak gusts since tornado intensities and associated wind speeds are based on observed damage. See WHA-B3
W-N-16	The high winds PRA failure calculations begin at a specified minimum V_L wind speeds. Wind speeds less than V_L are evaluated as not contributing significantly to SSC failure frequencies in the calculation of the plant's risk. Plant challenges resulting from wind speeds lower than V_L are assumed to be part of weather patterns that are implicitly included in the internal events PRA. For example, the internal event PRA uses a loss of off-site power (LOOP) frequency associated with weather phenomena at the plant and the electrical supply grid. The high winds PRA is therefore only concerned with high winds greater than V_L that strike the plant site. These winds may sometimes strike the grid away from the plant, but they must strike the plant to be considered in the scope of a high winds PRA. High winds PRA events remote from the site are generally considered within the internal events LOOP assessment and are included in the internal events PRA model via the LOOP initiating event frequency. Reviews of the plant's wind damage experience and the SSC design bases are suggested as part of the determination of V_L. The V_L wind speed should be low enough to capture the lower tail of fragility functions of the most vulnerable HWEL SSCs, but not so low as to include winds that are not risk-significant to the most vulnerable SSCs on the HWEL. Similarly, if V_L is too high, then the high winds PRA will ignore potentially important risk contributions from modest winds. For example, Kaasalainen et al. (2015) [W-25], Mironenko and Lovelace (2015) [W-26], and Kitlan and Mironenko (2016) [W-27] point out the dominant contributions of winds in the 73–157 mph range to the risk in recent high winds PRAs. A number of recent high winds PRAs used a V_L value of 73 mph, which corresponds to Fujita's original F1 wind speed range (Fujita, 1971 [W-28]). Starting at 73 mph may result in the loss of some of the lower tail contribution to failure of weak structures, such as a transmission tower (see Twisdale et al. 2015 [W-29]) or cladding from a metal clad structure, but a tradeoff is warranted to avoid having the dominant risk contribution be from random failures (versus high wind failures) in the lowest wind speed interval used in the high winds PRA calculations. In general, wind speeds lower than about 73 mph are assumed to be considered in the internal events model. Many plants have experienced maximum winds within 60–70 mph, which is consistent with the number of plant operating years and straight wind hazard analysis. Only a few plants have experienced winds over 80 mph. Typical design criteria for plant transmission lines as well as turbine building cladding often result in failure fragilities that are not insignificant for winds less than 100 mph (e.g., see Twisdale et al. 2015; Lovelace et al. 2017 [W-30]; Twisdale, 2017 [W-31]; and Banik et al. 2017 [W-19]). The selection of V_L remains an area where coordination with internal events PRAs is needed due to the potential sensitivity of the results to the selection of V_L. See WHA-B4

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-17	Wind hazard frequency curves are steep, typically characterized by a significant change in exceedance frequency for relatively small changes in wind speed. This characteristic influences the number and spacing of the wind speed intervals needed for accurate calculation of failure frequencies in the high winds PRA. Twisdale et al. (2015) [W-29] present results on how the number of discrete wind speed intervals in the computation of SSC failure frequencies impact the plant's computed plant risks. This paper showed that using too few intervals, especially for low wind speeds, results in overestimation of the failure frequencies. The plant risk was overestimated by about 35% when 5 intervals versus 10 intervals were used. The paper recommends at least 10 wind speed intervals for reasonably accurate failure frequency calculations in high winds PRAs. Kaasalainen et al. (2015) [W-25] similarly point out that the use of 10 versus 5 calculation intervals reduced the plant risk metrics by 50%. See WHA-B5
W-N-18	Although this note is with respect to LWRs, the risk-insights are important and may help inform non-LWR high winds PRAs. Recent LWR high winds PRAs (Kaasalanienan et al. 2015 [W-25]; Mironenko and Lovelace, 2015 [W-26]; Kitlan and Mironenko, 2016 [W-27]) indicate that the major contributions to risk metric frequencies for LWRs occur at wind speeds less than about 150 mph. Direct measurements of high wind speeds are difficult due to their rare occurrences, relatively small areas affected by the highest winds, and sparse network of measurement systems, some of which are also vulnerable to failure in extreme winds. Notwithstanding the potential limitations of high wind speed measurements, the following references give some indication of high observations to date: 150 mph for downbursts (Fujita, 1985 [W-32]), about 200+ mph for tropical cyclones (Hurricanes Camille, Patricia, Allen, Wilma, etc.), about 200+ mph for extratropical cyclones (Cerveny, 2007 [W-33]), and 300+ mph for tornadoes with mobile Doppler radar (Wurman, 2006 [W-34]). These observations, coupled with the exponential relationship between annual exceedance frequency (AEF) and wind speed, frequently make attempts to truncate or limit the wind speeds in the site wind hazard model unsuccessful with little potential benefit in terms of impact on the computed risk metrics. Twisdale et al. (2015) [W-29] presents failure frequency integration results for a range of 73–318 mph, with the last calculation interval covering a range of 260–318 mph. The last range was sufficient to capture the upper tail fragility contributions for all but the strongest SSCs. See WHA-B6

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-19	Most wind engineers rely on the wind speed data archived by the National Climatic Data Center. The user of the data should find information on anemometer height, anemometer type, averaging times, and exposure (surrounding terrain) and account for the effects of these in the analysis of the wind speed data. Data from multiple stations around the site are commonly used in straight wind frequency analysis (ASCE 7 [W-1], [W-2], and [W-3]; Vickery and Twisdale, 2014 [W-23]). Multiple stations within the same regional climatology provide significantly more data to ensure confident estimation of rare wind speeds, help to ensure that anomalous or poor quality data from any one station is not used in the analysis, and provide a spatial view of the regional straight wind frequencies near the site. Twisdale (2016) [W-35] notes that from five to eight regional NOAA stations have been used for the site analysis in recent high winds PRAs. For sites in complex terrain, the analysts should also check national wind standards and other publications to see if the site is in a "special wind" region. In ASCE 7, special winds refer to "regions in which wind speed anomalies are known to exist, such as winds blowing over mountain ranges, through gorges, or river valleys." As an example, the downslope winds near Boulder, Colorado, are a well-known special wind phenomenon (Durran, 2003 [W-36]). Wind speeds in these regions can be substantially higher than those indicated on the ASCE 7 wind speed maps. In the U.S., wind maps in ASCE 7 can generally be used in the determination of special wind region. In addition, knowledge of local meteorology conditions and historical storms may also be a part of assessing the potential for special winds at a site. Information from site anemometers, weather records, or other historical information may help determine whether special wind conditions are present. See WHA-C1
W-N-20	Operating nuclear plants typically have a meteorological tower with archived wind data. The analyst must evaluate the siting history and exposure of the anemometer, continuity of records, and type of archived data to determine whether it can be used for wind speed frequency analysis. Vickery and Twisdale (2014) [W-23] present a wind hazard frequency analysis based on a plant's archived hourly data, noting that the plant did not have archived peak gust data. A comparison of the wind speed frequencies from the site analysis (based on hourly data) with the surrounding NOAA stations shows a very significant underestimation of peak gust wind speed AEFs. These differences were judged to be due to the effects of terrain at the plant and the use of hourly averages, which effectively removes thunderstorm gusts. In this case, the site data was therefore rejected and not used in this operating reactor high winds PRA. See WHA-C1

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-21	Data used in the hazard analysis should be consistent in terms of the same averaging time, height, and open terrain conditions. Masters et al. (2010) [W-24] provide a reference for the adjustments for height, averaging times, and the effects of upstream terrain. Gust factor curves such as those given in ASCE 7 [W-1], [W-2], and [W-3] can also be used to account for small averaging time differences. In addition, adjustments for height, terrain, and averaging time can be performed using information given in the textbook by Simiu and Scanlan (1996) [W-37]. Also, Wieringa (1973) [W-38] and Beljaars (1987) [W-39] provide methods to adjust for terrain if information on the gust can be derived from the data. If gust data are not available, then the roughness of the local and upstream terrain can be estimated using aerial imagery, and mapping of the land use category to surface roughness using data published in the literature (e.g., Wieringa, 1992 [W-40]). The wind speed data in ASCE 7-16 [W-3] include such corrections for all the stations considered in the wind speed map development. Vickery and Twisdale (2014) [W-23] demonstrate the importance of understanding the details of the historical wind records at each meteorology station. Since wind measurement instruments and standards change over time, the averaging time, exposure and height of the anemometer, and type of anemometer may introduce important eras/biases into the wind speed frequency analysis. The sample frequency and averaging times of the measured wind speed must be obtained from the source providing the wind speed data. As noted in WHA-B2, if the wind speed measurements are periodic, it is likely that the wind hazard frequencies developed from such data will underestimate the true high wind hazard since the data will not contain the true daily or annual maxima. For advanced analysis, ESDU (2002) [W-41] provides a computer code for assessing the effects of upstream roughness on anemometer wind speeds. Other computer models are available that can be used to adjust for terrain. Judgment is usually required in the application of the terrain adjustment factors (e.g., ASCE 7-16). To the maximum extent practical, the analyst should ensure that computer programs employed in the analysis of wind data have the appropriate pedigree by virtue of being benchmarked against actual phenomena, and that they possess adequate validation/verification. See WHA-C2
W-N-22	A CC-I straight wind analysis provides for an extreme value analysis of anemometer station data without distinguishing the type of storm that produced the data. When storm types are not separated out, the resulting data are "mixed" and there will be data from large scale systems, like extratropical storms, as well as data from severe local storms, like thunderstorms. Using mixed data to predict rare, high wind speed events can introduce considerable unknown bias errors and uncertainties in the resulting wind hazard frequencies (e.g., see Holmes, 2015 [W-42]; Lombardo et al. 2012 [W-43]). A significant problem associated with CC-I can occur when annual extremes for a few decades of data are used to produce wind speed risk for return periods > 100 years. Since straight winds often dominate a site's wind hazard frequencies out to 1,000+ year return periods or more, the use of mixed data can introduce considerable errors and uncertainties. These problems can be reduced with the use of multiple stations of data, including the use of a superstation approach as was done for ASCE 7-98 [W-1] (Peterka and Shahid, 1998 [W-44]). Other methods include the method of independent storms (Cook, 1982 [W-45]) and peaks over threshold method (Davison and Smith, 1990 [W-46]). If the site is in a hurricane prone region, extremes from tropical cyclone storms should be removed from the mixed straight wind data set prior to analysis. See WHA-C3

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-23	Information is available in many countries to distinguish thunderstorm from non-thunderstorm straight winds, including, but not limited to, the following: the U.S., Canada, Germany, South Africa, and Australia. The separation of thunderstorm and non-thunderstorm winds and their treatment as statistically independent events was first proposed by Gomes and Vickery (1977) [W-47] for developing wind hazards in Australia. Vickery (1995) [W-48] demonstrated the applicability of the method in the U.S. Twisdale and Vickery (1993) [W-49] also showed that thunderstorms dominated the extreme winds over most of the inland U.S. Recent publications include Letchford and Ghosalkar (2004) [W-50], Lombardo et al. (2009) [W-51], and Lombardo et al. (2012) [W-43]. Vickery and Twisdale (2014) [W-23] summarize the approach and present discussions and methods for uncertainty analysis. Holmes (2015) [W-42] illustrates how the separation of thunderstorm from "synoptic" winds and their recombination produces a combined straight wind distribution that captures both the synoptic winds at less frequent return periods and the thunderstorm "downbursts" at more frequent return periods. A downburst is an area of strong, often damaging, winds produced by one or more convective downdrafts (AMS, 2013 [W-52]). Downburst wind speeds of 150 mph have been measured at an airport and other estimates of straight wind gusts up to 179 mph reported (e.g., Fujita, 1985 [W-32]; NOAA, 2002 [W-12]). The separation methodology is the basis for the straight wind analysis used in the wind speed maps in ASCE 7-16 [W-3], noting that the ASCE maps also include hurricane winds near the coast. Straight wind speed maps based on separation of thunderstorm and non-thunderstorm winds (without hurricanes) are given in map form in Pintar et al. (2015) [W-53]. The benefits of a separate analysis are more accurate estimations of extreme straight winds and reduced uncertainties resulting from having larger data sets. For example, if a station has 30 years of reliable data and there are an average of 15 thunderstorms a year, a total of 450 wind speed events can be used in the analysis for a single station, whereas an annual extreme value analysis would only have 30 wind speeds (of mixed events). The use of coherent (not mixed) data sets with many more events provides much more confidence for the important wind speed AEFs $< 10^{-2}$. The state-of-the-art and practice for combining different wind hazard type (such as thunderstorms and extratropical cyclone) frequencies on a per year time interval is to assume statistical independence (e.g., Simiu and Scanlan, 1996 [W-37]; Vickery and Twisdale, 2014 [W-23]; ASCE 7, 2016). See WHA-C3
W-N-24	As a point of reference, a commonly used wind speed distribution for straight wind analysis is the Gumbel or Extreme Value Type I Distribution. For example, ASCE 7 [W-1], [W-2], and [W-3] have used the Type I for previous versions of this Standard. Some researchers have investigated "tail-limited" distributions (e.g., Simiu and Heckert, 1998 [W-54]). Tail limited distributions are strongly influenced by a few wind speeds in the tail of the distribution. A major concern with tail-limited distributions in a high winds PRA risk assessment is the capping of wind speeds based on a limited data set that may not include rare but intense downburst winds that may dominate straight wind AEFs $< 10^{-2}$. That is, a 20- or 30-year data set is unlikely to include rare, small-scale straight wind phenomena associated with, for example, 100, 500, 1,000, and 5,000+ year return periods. The concern is that wind speeds for this range of return periods cannot be accurately estimated by tail-limited distributions based on data samples that do not include such phenomena. The use of tail-limited distributions has been questioned by, for example, Cook and Harris, 2001 [W-55]; Harris, 2005 [W-56]). Thus, if the analyst uses a tail-limited distribution, it is recommended that a supporting basis be developed and documentation be provided to support a technical peer review. See WHA-C4

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-25	In the analysis of straight winds, multiple stations are often used due to the limitations of short data records for any one station. For a nuclear plant site, the analyst must often determine how to combine the regional data/analyses to produce the site-specific straight wind risk. For example, the analyst might make the case for equal weights in a homogeneous region; or, he/she might conclude that weights should be based on the inverse of distance from the plant. A superstation approach could also be considered, as was used by Peterka and Shahid (1998) [W-44] in ASCE 7-98 [W-1]. ASCE 7-16 [W-3] used a smoothing approach from a large number of individual stations in the updated wind speed maps. Due to the aforementioned issues on data quality, data corrections, limitations of short-term records, and large uncertainties, the use of a single station's records for a site's straight wind model should be avoided when lacking state-of-practice regional analyses/comparisons to provide adequate justification. See WHA-C5
W-N-26	Comparison of the straight-wind hazard frequencies developed for the Wind Hazard Analysis to data specified in national wind loading standards (or to relevant data published in the literature following publication of the national standard) provides useful perspectives for both the analyst and the peer reviewers. Wind hazard curves used in wind loading standards (and most of those that appear in the literature) do not include the effects of propagating epistemic uncertainties. In general, due to the non-linear nature of error propagation, the inclusion of epistemic uncertainties produces higher mean hazard frequencies (NUREG/CR-6372 (Budnitz et al. 1997) [W-7]; and Vickery and Twisdale, 2014 [W-23]). As discussed by Vickery and Twisdale (2014), aleatory parametric uncertainties typically include the statistical parameters of the extreme value distribution. Epistemic uncertainties may include, among others, corrections for anemometer height, which is based on surface roughness; corrections for surrounding terrain; and corrections for averaging time based on measurement systems used in the data record period. Vickery and Twisdale (2014) demonstrate how the combination of regional extremes in a site analysis result in uncertainty curves that contain rare extreme straight wind speeds, whereas, by comparison, the same data appear as outliers to a specific site's 5^{th} and 95^{th} uncertainty curves. See WHA-C6
W-N-27	In recent high winds PRAs, two methods have been applied for the tropical cyclone wind speed frequency analysis. *(a)* **Existing Study:** Examples of tropical cyclone publications that provide sufficient information to derive a hazard curve include Vickery et al. (2009a) [W-57], coupled with NUREG CR/7005 (Vickery et al. 2011) [W-10], ASCE 7-10 [W-2], or ASCE 7-16 [W-3]. It is important to note that the latter two examples contain both tropical cyclone and non-tropical cyclone data combined as statistically independent processes. *(b)* **Model Calculations:** Tropical cyclone wind hazard curves cannot be developed using historical wind speed data. A tropical cyclone simulation model is the preferred method for developing the wind speed frequencies. For use in a High Winds PRA, the model should be published and represent the current state of the art. Each model component should be individually validated. Such model components include, but are not limited to, the wind field model, statistical models for storm size, central pressure, frequency, landfall location, translation speed, heading, and Holland B (Holland, 1980 [W-58]) or other parameters that control the relationship between central pressure and wind speed. Examples of model validation are given in Vickery et al. (2000 [W-59], 2009 [W-60]) and James and Mason (2005) [W-61]. A discussion of hurricane hazard modeling is also contained in Vickery et al. (2009a). See WHA-D1

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-28	The data sources listed in WHA-D1 commentary provide several sources that can be used to compare results from model calculations performed under WHA-D1. There are aleatory and epistemic uncertainties in each component of a tropical cyclone model. For the model calculation approach, the uncertainties in the data and in each sub-model should be propagated through the hazard model. Regarding data sources, it is important to note that the period of record of quality tropical cyclone data varies from basin-to-basin and with the type of data. For example, along the U.S. coastline, there is an almost complete set of landfall central pressure, heading, translation speed, landfall location, and frequency data extending back to about 1900, whereas quality data pertaining to storm size are limited to the 1940s and later. Maximum wind speeds given in the historical databases are estimated values rather than measurements. See WHA-D2
W-N-29	Recent publications reinforce long-recognized tornado data issues and indicate a number of concerns in tornado risk estimation that were not considered in the above referenced national tornado map products, including under-estimation of risk due to unreported events in the modern era (Elsner et al. 2013 [W-62]; Skow and Cogil, 2017 [W-63]); low-biased wind speeds in the Expanded Fujita scale (Twisdale et al. 2016 [W-64]); potential underestimated EF ratings versus Doppler radar measurements (Wurman and Kosiba, 2014 [W-65]); path length intensity variation data (Faletra et al. 2016a [W-66]); uncertainties in damage-based tornado ratings (Edwards et al. 2013 [W-67]); and significant biases/errors in the evolution of the U.S. national tornado databases (Verbout et al. 2006 [W-68]; Faletra et al. 2016b [W-69]). The resources mentioned above and many others provide insights for model-based calculations of tornado hazard risk. With regard to new research and information sources, it is important to note that significant efforts are underway to develop ASCE standards for tornado wind speed estimation (LaDue, 2016 [W-70]), tornado hazard maps (Phan et al. 2016 [W-71]), and tornado design standards for structural design (Kuligowski et al. 2014 [W-72]). These efforts are expected to improve considerably on the current vintage of tornado wind speed risk maps and provide a significant new resource for high winds PRA. See WHA-E1
W-N-30	As discussed in WHA-E1, there is a significant and growing literature on tornado hazard modeling regarding the technical requirements in WHA-E2. The references listed in WHA-E1 provide a good primer for reviewing the important issues and variables influencing the analysis of tornado wind frequencies. A major consideration is that the EF scale wind speeds may under-represent tornado winds and are undergoing critical analysis for use in ASCE tornado wind speed map estimation (LaDue, 2016 [W-70]). With the many data limitations and assumptions required in tornado hazard analysis, epistemic uncertainties are an important part of the tornado hazard modeling and analysis. See WHA-E2
W-N-31	The effects of target size in tornado hazard analysis have been documented in the literature since the 1970s (e.g., Garson et al. 1975 [W-73]; Wen and Chu, 1973 [W-74]; and Twisdale and Dunn, 1979 [W-75]). As the target size becomes larger (for example, a tornado striking any SSC at a nuclear plant), the chance of a tornado striking the target increases. This well-established effect creates a unique linkage between the tornado hazard curve development and its use in the fragility analysis. Twisdale et al. (2015a) [W-76] illustrate typical tornado hazard curves for a point target (e.g., small building) and an example nuclear plant target used in wind-generated missile analysis. The ratios between these curves typically range from about 2 to 3 at low wind speeds to factors of 10 or more at high wind speeds. See WHA-E4
W-N-32	Due to the aforementioned complexities of the tornado databases and wind speed estimation, the analyst should make a careful evaluation of the potential limitations of past standards with respect to up-to-date publications in any tornado hazard comparisons. See WHA-E5

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-33	Examples of aleatory and epistemic uncertainties and references are given in the discussion for the respective wind hazards. See WHA-F1
W-N-34	Examples of fractile and mean hazard curves for tornadoes are provided in Twisdale et al. (2015a) [W-76] and Twisdale (2016) [W-35]. Examples of the 5^{th}, 95^{th}, and derived mean for straight winds and hurricanes are given in Vickery and Twisdale (2014) [W-23]. See WHA-F3, WHA-F4
W-N-35	This SR is not applicable to operating plants. See WHA-G3, WFR-A4, WFR-I3, WPR-F3
W-N-36	This SR is not applicable to PRAs performed for a specific site. See WHA-G4, WFR-A1, WFR-B5
W-N-37	For PRAs performed on a bounding site, the intent is for the fragility analysis to be decoupled from the hazard analysis to the extent practical to allow for the fragility analysis to bound a range of sites. As such, generic data within Wind Fragility Analysis should be used as much as possible. See WFR-A1, WFR-A3, WFR-A5, WFR-A6, WFR-A7, WFR-A8, WFR-A9
W-N-38	Although a site-specific and plant-specific analysis of the fragilities of SSCs provides the most realistic estimates, for the pre-operational stage PRA, it may be acceptable to develop fragility estimates on some other basis (e.g., generic information), provided that the uncertainties introduced are acceptable for the application. For PRAs performed during the pre-operational stage, decoupling the fragility analysis from the hazard analysis to the extent practical allows for the fragility analysis to bound a range of sites. It may be necessary to use generic data for this purpose. See WFR-A4, WFR-B2, WFR-B3, WFR-B7, WFR-C1, WFR-C2, WFR-C3, WFR-C4, WFR-F1
W-N-39	SSCs can fail under different wind effects. A wind effect, such as wind pressure or APC, can result in multiple failure modes, such as overturning, shear, bending, tension, uplift, etc. The analyst should identify the dominant effects and failures modes for each SSC, considering the potential for both simultaneous-in-time effects and separated-in-time effects. Wind pressure loading effects are often considered with respect to a main wind force resisting system and components and cladding (C&C) (see ASCE 7-16 [W-3]). Missile impact effects are often considered with respect to local effects (such as penetration, perforation, spall) and overall effects (crimping, bending, shear, and other structural or support failures). A common way to communicate these considerations is through a master list table that enumerates all the failure modes for each SSC. See WFR-A5
W-N-40	Wind fragility analysis is complicated by the presence of multiple wind effects and the potential for multiple failure modes for each wind effect. In general, aggregations of effects and failure modes are essential in a Wind Fragility Analysis. A systematic mapping of the wind effects, potential failure modes, and structural interaction to each SSC provides a reasonable approach to help organize the analysis, the aggregation, and the rationale. For wind pressure fragilities, the use of a code-based approach (Kennedy and Ravindra, 1984 [W-77]; Twisdale, 2017 [W-31]) can be used to simplify the assessment of failure modes for complex structures by using load and resistance factors. As discussed by Lovelace et al. (2017) [W-30], area/volumetric modeling can be an effective method to handle the complexities associated with numerous equipment items within an area or room. This approach is similar to the "rule of the box" approach used in seismic studies (Lovelace et al. 2017a [W-14]). See WFR-A6

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-41	Using the same fragilities for different wind hazards depends on the similarities, or lack thereof, of the characteristics of the wind hazard. Tornado winds are generally viewed as significantly different as non-tornadic winds. For example, tornado winds include a vertical wind component and APC load effects. The rotational wind component in a tornado is often on a scale that is similar to the building length or width dimension, producing high winds from multiple directions in the same event. The tornado wind-generated missile risk has different characteristics than straight wind or hurricane wind missiles. Hence, recent high winds PRAs have evolved to use separate fragilities for tornadic and non-tornadic winds (Twisdale, 2016 [W-35]). An important simplification for non-tornadic fragilities results from similarities in gust factors and velocity profiles. For example, Vickery and Skerlj (2006) [W-78] showed that the gust factors and velocity profiles in hurricanes can be treated with the standard factors used for extratropical storms, which is the current basis for the pressure loads in ASCE 7 [W-1], [W-2], and [W-3]. While there are important differences in thunderstorm wind characteristics (Twisdale and Vickery, 1993 [W-49]), there is insufficient data on which to develop separate thunderstorm wind load effects to warrant separate fragilities for these short duration hazards. Hence, recent high winds PRAs have simplified the calculations to the following: *(a)* tornadic fragilities specifically for the tornado hazard; and *(b)* non-tornadic fragilities for straight winds and hurricanes. Regarding wind-driven rain fragilities, several recent high winds PRAs have shown that the rain probability and intensity are hazard dependent; for example, there is a much higher risk of intense rain during tropical cyclone events than thunderstorms or tornadoes. See WFR-A7
W-N-42	The goal of this SR is to identify and evaluate potentially significant correlations for wind effects and failure modes. This requirement is listed under HLR-WFR-A to avoid repetition under each wind effect SRs in HLR-WFR-D through HLR-WFR-G. The potential wind-effect and failure mode correlations are numerous, difficult to judge, and could require complicated 3-D physical models to accurately estimate. The degree of correlation is likely to depend on the wind speed interval. The amount of failure mode correlation at low wind speeds will likely be different than the correlation at high wind speeds. Correlation of failures across SCCs within the same wind event and wind speed interval may also vary significantly from one SSC to another. SSCs that are separated physically, are not in the same structure, and have opposite wind direction vulnerabilities may have negatively correlated failures. Structural interactions from failures of a building frame, tower, etc., may produce positive correlations across SSCs within or near these structures. A piping system could fail by missile perforation or crimping of the pipe, and these failure modes may be independent, or positively, or negatively correlated, based on the wind speed interval. A discussion on wind fragility correlations and the development of simple correlation bounds following structural reliability concepts are given in Twisdale, et al. (2015) [W-29]. It is important to note that the simple bounds in this paper are based on percent differences (and not ratios) with respect to a baseline assumption of statistical independence. Tighter bounds on SSC failure mode correlation can be obtained through more detailed approaches (e.g., see Ditlevsen, 1979 [W-79]). In summary, due to (a) the complexity of wind fragility correlation analysis; (b) the lack of published research/data in this area; and (c) the potential for a large number of impacted SSCs in a high winds PRA, engineering judgment coupled with the use of bounding correlation assumptions, and sensitivity analyses may be useful for evaluating important fragility-related correlations. In recent high winds PRAs, SSCs that are in close proximity or are exposed to structural interactions have been viewed as important correlations that require consideration. For example, extremely close SSCs may be vulnerable to damage from a single large missile through simultaneous impact or missile ricochet. See WFR-A8
W-N-43	Coexistent hazards include wind-driven rain and potential flood surge. See WFR-A9, WPR-B8

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-44	The action verb ADDRESS does not imply the need for quantitative analyses. See WFR-A9
W-N-45	See Sciaudone et al. (2015) [W-13] and Lovelace et al. (2017a) [W-14] for investigation insights and guidance for high winds PRA. Also see EPRI, "High Wind Equipment List and Walkdown Guidance" 3002008092 [W-15] for additional guidance and as a comprehensive general resource (including example photographs from high wind walkdowns conducted at operating nuclear power plants) for conducting a high winds PRA investigation. See WFR-B1
W-N-46	See Lovelace et al. (2017a) [W-14], Sciaudone et al. (2015) [W-13], and Twisdale et al. (1978 [W-80], 1981 [W-81]) for missile source modeling discussions and missile walkdown guidance. It is important to note that wind-generated missiles from failed structures are a major source of missiles in high winds PRAs. Cladding and roof deck failures are an important source of missiles. Roof ballast, such a pavers, and gravel, may be significant since these potential missiles can start at a high elevation, such as the roof and walls of a turbine building. Twisdale (2016) [W-35] noted that the number of missiles developed through recent plant surveys included over 150,000, of which > 50% were from failed building sources. Metal cladding missiles have been identified to be a major source of risk at several plants. See WFR-B3
W-N-47	This SR is not applicable to PRAs performed on a bounding site. See WFR-B4, WFR-E9
W-N-48	The plant operating conditions at the time of the missile survey are important. Outages generally result in a significant number of additional missile sources, including trailers, close to many SSCs (Sciaudone et al. 2015 [W-13]). A modeling approach to account for plant operating conditions is illustrated by Twisdale (2016) [W-35]. The buildup of materials prior to the outage, the outage duration, and the post-outage cleanup can amount to a notable fraction of time from one outage to another, particularly for multiple reactors. See WFR-B7
W-N-49	Screening for wind pressure effects could include, for example, seismic Category I structures that were designed for 360 mph tornado winds and APC effects. The justification could be based on a code-based analysis with load and resistance factors. Another example of wind pressure screening is a simplified overturning analysis of a tank or large piece of detached external equipment. The analyst might use bounding, conservative assumptions on overturning load and weight to estimate a conservative, over-turning wind speed. This type of screening would eliminate the need for more detailed fragility modeling for overturning failures. As part of the screening process, it is important to evaluate doors, vents, and other components for their design basis and/or evaluate if failure of these components would affect interior, safety-related SSCs. A high wind plant-design basis may not always suffice for screening. For example, a concrete roof slab of a seismic Category I structure may only be 8 inches thick and vulnerable to missile induced spall or perforation. Twisdale et al. (1981) [W-81] include information on missile perforation and spall of reinforced concrete that may be useful for screening. In addition, a plant's missile design basis may not be sufficient for screening for missile effects. For example, a plant may have a wood beam and an automobile missile as the "design basis" missiles, with the automobile missile trajectory height limited to 30 ft. Such missile design bases do not eliminate risk from the steel cladding, purlins, girts, pipes, and other structural shapes that may become missiles at a plant and impact SSCs at all elevations. High winds PRA screening of SSCs results in assigning a fragility of 0 to those SSCs not susceptible to high winds. The SSC, such as a pump or motor, can still fail from random failures, and, therefore, it should not necessarily be excluded from the high winds PRA, even if it is screened from high winds PRA failures. See WFR-C1

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-50	SSCs that are screened for missiles should not have significant risk vulnerabilities from missiles passing through doors, louvers, vents, or other openings. There is little guidance to determine significant missile risk vulnerabilities. "Line of sight" from the opening to the SSC is an approach that has been used within the industry. However, plant walkdowns have also uncovered situations where there may not have been a "line of sight" vulnerability, but where a missile ricocheting from a nearby concrete surface is re-directed toward and into the SSC (e.g., Sciaudone et al. 2015 [W-13]; Twisdale et al. 2015 [W-29]; Twisdale, 2016 [W-35]). Also, if a wind-generated missile enters a building at a high elevation and then falls through the building, ricocheting along the way, a horizontal line-of-sight assessment would not be an acceptable screening approach. See WFR-C2
W-N-51	SSCs that are qualified for outdoor environments and associated rain deposition and wind-driven rain should qualify for screening. NUREG/CR-6533, SAND97-1735 (Murata et al.1977) [W-8] provide classes of electrical equipment that may be useful in evaluating the potential vulnerability to wind-driven rain. See WFR-C4
W-N-52	In the U.S., ASCE 7 [W-1], [W-2], and [W-3] is an acceptable national wind standard for wind pressure loading for non-tornadic winds. In modeling wind pressure load effects, an important consideration is the role of internal pressures, which are considered with respect to an open, enclosed, or partially enclosed building (ASCE 7). In windstorms, buildings may fail progressively (cladding, doors, vents, roof elements, etc.), and the enclosure state may change from enclosed to partially enclosed to open. The enclosure state affects the internal pressures and net loads on the building. The role of progressive failure is well documented in the literature (e.g., Minor et al. 1977 [W-82]; Vickery et al. 2006 [W-20]; Twisdale, 2017 [W-31]; and Banik et al. 2017 [W-19]). See WFR-D1
W-N-53	Tornadoes are capable of producing significant APC loads (e.g., Simiu and Scanlan, 1996 [W-37]; Roueche, 2017 [W-83] . The effects of APC loads are related to the background building leakage, envelope failures, the size of the tornado radius of maximum winds (RMW) relative to the building, horizontal wind speed, translational speed, and other factors. Due to the limitations of the state-of-the-art in modeling APC, simplified approaches with considerations of epistemic uncertainties may be appropriate. See WFR-D2
W-N-54	In the code-based approach (Kennedy and Ravindra, 1984 [W-77]) that is often used for wind pressure fragility analysis, factors are developed based on load and resistance information. Twisdale et al. (2015a) [W-76] and Twisdale (2017) [W-31] discuss an enhanced code-based approach for wind pressure fragility analysis for structures and building envelopes. Regarding the use of a code-based approach, it is important to note that the loads and resistances referenced in earlier code eras may be significantly different than those employed today. For example, if a building was designed in the 1960s or 1970s, the pressure coefficients and reference wind are not the same as used in modern code. The analyst should identify and correct for the important differences in the fragility analysis. Again, in the past, certain standards used fastest-mile wind speeds. Other standards simply allowed for a one-third stress increase to account for wind loads, which is no longer permitted. Pressure coefficients on C&C have increased in recent codes, incorporating improved wind tunnel data. These code differences can be important in an analysis of wind pressure fragilities for SSCs built prior to the most recent editions of the national building loading standard code. See WFR-D3
W-N-55	The analyst must determine whether an SSC is a rigid or flexible (such as a chimney or tall building) structure. If the structure is flexible, the dynamic response characteristics should be included in the fragility analysis. For reference, ASCE 7 [W-1], [W-2], and [W-3] defines a rigid structure as one with a fundamental frequency greater than or equal to 1 Hz. See WFR-D4

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-56	A procedure for topographically induced wind speed-up effects is provided in ASCE 7 [W-1], [W-2], and [W-3], including background references. It is important to note that Section 26.8 in ASCE 7-16 [W-3] does not address the general case of "wind flow over a hilly or complex terrain for which engineering experience, expert advice, or wind tunnel procedure may be required." See WFR-D5
W-N-57	Code pressure coefficients in standards such as ASCE 7 [W-1], [W-2], and [W-3] are based on isolated building wind tunnel tests. The effects that groups of buildings can produce on each other result in "shielding or negative shielding" effects on an individual structure. Shielding and negative shielding are also referred to as sheltering and negative sheltering (Cook, 1990 [W-84]) and shielding and channeling (ASME/ANS, 2013 [W-85]). Others, such as Ho et al. (1990) [W-86], refer to these same effects as "variability of wind loads due to obstructions of surrounding buildings." Shielding effects are well recognized in wind engineering and provide the rationale for wind tunnel testing of tall buildings and bridges throughout the world. As implied, the effects of shielding and negative shielding can either reduce the loads or increase the loads on a structure. Based on wind direction, it is possible to increase the loads without a clear channeling set-up of the surrounding buildings. This SR directs that the analyst consider the potential for shielding and negative shielding in the wind pressure fragility analysis. One possible way to do this is through the use of statistical factors based on wind tunnel tests that introduce variability in the loads (Ho, et al. 1990). The data can allow for a statistical treatment of both shielding and negative shielding (channeling of winds between obstructions that can produce speed ups). In summary, the nature of nuclear plant sites, typically with groups of buildings located near the center of the site, will likely result in shielding and negative shielding effects. Without a wind tunnel test, the state-of-practice is to apply factors/engineering judgment to account for these effects. See WFR-D6
W-N-58	It is important that site-specific hazard models and data be used in the development of wind-generated missile fragilities. Missile fragilities are based on missile impact and damage assessments. Missile fragilities therefore include the probability of one or more missiles impacting the SSC as well as the probability of damage to the SSC. Both missile impact probability and damage given an impact are dependent on the missile type. Lightweight missiles may have a much higher impact probability than heavy missiles. However, lightweight missiles may not be able to damage certain rugged SSCs. Thus, it is important to develop missile fragilities in a manner that recognizes the missile type dependence in the fragility analysis of the SSC. Important hazard characteristics include the velocity profile of the horizontal winds, storm width, storm direction, rotational velocity components, RMW, and vertical winds. These hazard characteristics are different for tornadic and non-tornadic wind hazards. Recent high winds PRAs have analyzed missile risk for two major classes of wind hazard: tornadic and non-tornadic (Twisdale, 2016 [W-35]). Non-tornadic winds include straight winds and tropical cyclones. The wind hazard strike definition should include a broad area around the SSCs since missiles can be generated and transported significant distances (see commentary for WFR-E6, below). Twisdale et al. (1978 [W-80], 1981 [W-81]) and Sciaudone et al. (2015) [W-13] illustrate missile generation areas for plants. Twisdale et al. (2015a) [W-76] illustrate the difference in tornado hazard curves for a small building at a site versus broader plant-wide strikes for missile analysis. It is important to note that the tornado hazard data in NUREG/CR-4461 (Ramsdell and Rishel, 2007) [W-6] are for individual buildings and not site-wide missile analysis. See WFR-E1

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-59	Wind hazard characteristics are important in wind-generated missile fragilities. Simiu and Potra (2011) [W-9] conclude that average missile speeds are considerably higher in hurricanes than for tornadoes. Twisdale (2016) [W-35] also indicated significant differences in straight wind versus tornadic missile wind fragilities with straight winds producing higher fragilities in about 70% of the SSCs. The ratio differences between the two, on a target by target basis, exceeded factors of 10 for some SSCs. Straight winds cover a larger area and produce more missiles, on average, per event. Tornado winds can produce missiles that can reach high elevations, travel extreme distances, and achieve high velocities. Partial wind directional shielding of an SSC may not reduce the impact of tornado missiles but could impact non-tornadic wind missiles. Hence, although straight winds and tropical cyclone winds cover larger areas of the site, the results to date suggest that there is no single wind hazard that can be used conservatively to produce wind-generated missile fragilities for all wind hazards. In recent high winds PRAs, the analysts have developed separate missile fragilities for tornadic and non-tornadic hazards (Twisdale, 2016). Non-tornadic hazards may include both tropical cyclones and straight winds. In summary, given published results detailing missile fragility dependence on wind hazard, this SR requires justification to substitute missile parameters derived from one wind hazard, say tornadoes, for those from another wind hazard, such as tropical cyclones. See WFR-E2
W-N-60	There are many distinct missile types at a nuclear plant site. For example, see Twisdale et al. (1978, 1981) [W-80] and [W-81] regarding the need for a broad spectrum of missiles for probabilistic risk analysis of missile effects. Recent papers (Sciaudone et al. 2015 [W-13], 2017 [W-87]; Banik et al. 2017 [W-19]; Twisdale, 2016 [W-35]; and Northrup et al. 2017 [W-88]) emphasize the important role of structure source missiles, such as metal cladding. See WFR-E3
W-N-61	Structure source missiles may be one of the most important missile sources at a plant. Roof materials and cladding are elevated above the ground surface, exposed to higher wind speeds, and have further to fall to reach ground level when transported than missiles that originate near the ground. Progressive failure of the building envelope (Banik et al. 2017 [W-19]) provides a ready source of elevated missiles that can transport significant distances and at high speeds. It is important to note that the calculations and results in Twisdale et al. (1978, 1981) [W-80] and [W-81], which have been used in simplified model development (Hope et al. 2015 [W-91]; NEI 17-02, 2017 [W-89]), did not include significant structure missile sources. Current analyses that consider structure source missiles indicate that metal roof and wall cladding and associated purlins and girts, and occasionally roof pavers, can provide a significant source of missile risk at nuclear plants. See WFR-E4
W-N-62	The purpose of WFR-E5 is to provide justification for the missile source distance used in the missile fragility analysis. Based on sensitivity analyses, an exclusion distance of 2,000 to 2,500 ft. to cover the area of risk significance for tornado missiles was suggested in Twisdale et al. (1978) [W-80]. A number of recent high winds PRAs have used a 2,500 ft. exclusion distance from the nearest SSC for sites characterized by small elevation changes. Use of reduced distances may be possible with appropriate analysis and justification. Sites with significant elevation changes and missile sources at high elevations compared to the plant may require an enhanced missile source distance determination. See WFR-E5

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-63	Attempts to scale tornado missile probabilities include work by Reed and Ferrell, 1987 (NUREG/CR-4458) [W-5] and NEI 17-02 (2017) [W-89]. Scaling refers to approaches that attempt to use results for one set of plant-specific targets to estimate the missile probabilities for other targets with different areas and missile exposures. There are concerns regarding scaling of missile fragilities for targets of different sizes and exposures. For example, RIS 2008-14 (NRC 2008) [W-90] expressed concerns regarding improper scaling of area ratios. Sciaudone et al. (2017) [W-87] provide examples where statistical scaling of results produces errors that can exceed factors of 100 or more for individual plant SSCs. When scaling methods are used, large random and epistemic uncertainties may be generated that need to be evaluated to capture the potential for large errors for individual targets. Certain models (NEI 17-02, 2017 [W-89]; Hope et al. 2015 [W-91]) in the recent literature use original TORMIS data (Twisdale et al. 1978 [W-80], 1981 [W-81]) for scaling and/or validation. Care is needed in using dated TORMIS results in light of RIS 2008-14 (NRC, 2008) and the fact that the 1978-1981 TORMIS examples focused on large targets, were limited in the number of simulations performed, and included only a limited number of missiles compared to those used in recent high winds PRAs (including not accounting for structure source missile populations, nor a full spectrum of missiles in some cases). Recent high winds PRAs show that structure source missile populations can be significant sources of plant missiles contributing to missile risk (e.g., Twisdale, 2016 [W-35]). Sciaudone et al. (2017) [W-87] provide detailed discussion of several simplified models and produce comparative statistics. Large aleatory and epistemic uncertainties may be needed for evaluation when models that rely on scaling approaches are utilized. See WFR-E6
W-N-64	Due to the large number of missiles at most sites and the fact that a single missile may damage an SSC, a probabilistic missile analysis should ensure that the results capture the potential for any one missile to produce damage. For example, if there are 100,000 missiles at a site, 1 missile out of 100,000 is 10^{-5} of the missile distribution function. Hence, this requirement emphasizes the need to demonstrate stable probabilistic results considering the potentially large number of missile sources. For example, Twisdale et al. (2015a) [W-76] and Twisdale (2016) [W-35] show tornado missile analysis results and probabilistic convergence plots using a replication approach to quantifying the standard error in the mean fragilities. The RIS 2008-14 (NRC 2008) [W-90] emphasized the need for convergence in performing missile risk analysis. See WFR-E7
W-N-65	CC-I of WFR-E8 is the same as CC-I of WFR-E9. Only CC-II requirements differ to allow for less site specificity for bounding site analysis. See WFR-E8, WFR-E9
W-N-66	Wind generated missile effects at a plant are highly dependent on the analysis assumptions and the methodology components. The analyst should specify the assumptions and methods for the components listed. There are many publications in the area of wind-generated missiles for tornadoes and other wind hazards. For example, see Tachikawa (1983) [W-92], Lin et al. (2006) [W-93], Kordi and Kopp (2009) [W-94], Crawford (2012) [W-95], Hope et al. (2015) [W-91], and NEI 17-02 (2017) [W-89]. Sciaudone et al. (2017) [W-87] compare missile impact probabilities for several recently developed models used in the nuclear industry. See WFR-E8, WFR-E9

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-67	The additional CC-II requirements include site-specific hazard and plant geometry components that may be important in the quantification of missile effects. CC-II requires the use of missile dependent aerodynamics in the trajectory analysis and that missile impact and damage effects be missile type dependent. CC-II ensures that the analysis method captures the 3D spatial features of missile sources, trajectories, shielding and ricochet, and SSC locations. Benefits of the CC-II analysis are improved accuracy and reduced uncertainties in the missile effects analysis. For example, if the CC-II method has validated components, then the use of validated components may be important in the epistemic uncertainty modeling. Validation includes consideration of elements such as the numbers of missiles from source structures, the missile injection model, the trajectory model, and the damage model. As an illustration of wind-generated missile validation for missile, injection, trajectory distances, and missile damage modeling, see Twisdale et al. (1978 [W-80], 1979 [W-75], and 1981 [W-81]) and FEMA (2007) [W-96]. Experimental validation and sources of model data are also illustrated by Crawford (2012) [W-95], Lin et al. (2006) [W-93], and Kordi and Kopp (2009) [W-94]. See WFR-E8, WFR-E9
W-N-68	The distinction between CC-I versus CC-II missile effects analyses include the following: *(a)* consideration of site-specific shielding structures; *(b)* consideration of site-specific missile ricochet into SSCs, if appropriate; *(c)* use of site-specific wind hazard path size and path directions; *(d)* use of missile type-dependent aerodynamics; *(e)* use of missile damage methods that depend on missile type; *(f)* ensuring that the analysis components capture the site-specific and risk-significant 3-D features of the SSCs and plant. See WFR-E8
W-N-69	The missile hit/damage criterion should be described for each vulnerable SSC. For example, missile hit might be used for fragile SSCs or as a conservative criterion for all vulnerable SSCs. Penetration, perforation, spall, crimping, and other criteria might be used for SSCs with some degree of hardness. Twisdale et al. (1978 [W-80], 1981 [W-81]) discuss these effects. Bochierri et al. (2009) [W-97] and Northrup et al. (2017) [W-88] present detailed results from finite element calculation on nuclear plant SSCs. See WFR-E10
W-N-70	Correlated failures of SSCs are an important concern in PRAs. For example, damage to redundant components in a high winds PRA event may be a major contributor to plant risk. It is important to identify SSCs that may be subject to correlated missile failures. Close proximity without protection from missiles might indicate a potential positive correlation to missile damage in the same wind event. Hence, this SR requires a description as to how the missile fragility analysis considers the potential for multiple SSC failures from wind-generated missiles and why those failures are not statistically dependent. Correlation of missile fragilities may be important in the high winds PRA since many SSCs may be impacted by multiple missiles generated in a wind event. Twisdale et al. (2015) [W-29] discuss positive and negative missile fragility correlations and provide some simple correlation bounds. See WFR-E11
W-N-71	The missile populations at a plant vary over the lifetime of the plant. New structures are built, modifications are made, materials, additional vehicles, and temporary structures and offices are needed for outages. In some cases, significant amounts of materials may be stored near safety-related SSCs. Discussions of the importance of treating these missile population variations can be found in Sciaudone et al. (2015) [W-13] and Lovelace et al. (2017a) [W-14]. See WFR-E12

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-72	Structural interactions occur when the wind response of an SSC affects the response of other SSCs. For example, the collapse of the roof deck or roof structure could be assumed to fail all the SSCs located underneath. Structural interactions need to be considered based on location of SSCs within structures and the potential for a structure to collapse onto an adjacent structure. Recent high winds PRAs have demonstrated the need for the fragility modeling team to have a clear understanding of the relationship between structural interaction and wind-generated missile fragilities to (a) avoid overlooking potential structural interactions and (b) to ensure consistent modeling of both structural interaction fragilities and missile fragilities. For example, consider the potential failure of metal wall cladding. If vulnerable SSC "A" is located adjacent to the wall cladding, a potential structural interaction failure mode may include damage to "A" during the time that the cladding is becoming fully detached from the structural frame. That is, dynamic motions of the cladding element, while still partially attached to the wall, could produce repeated impacts on a nearby, vulnerable SSC. In this case, ignoring the interaction potential would underestimate the fragility of SSC "A." The cladding element may also become fully detached during the storm and transport as a missile, which could potentially hit SSC "B," located some distance away. See WFR-F1
W-N-73	High winds are often, but not always, accompanied by rain. Wind-driven rain is a potential wind effect that results when high winds produce damage to the plant and rain occurs during or shortly after the occurrence of the damaging winds. Wind-driven rain includes rain that has a horizontal velocity component from wind. This effect is potentially important when electrical equipment vulnerable to water damage is housed in structures whose building envelopes may fail in high winds. High winds could damage the roof cover, roof deck, or wall cladding, allowing wind-driven rainwater to enter the building and saturate the equipment (Twisdale et al. 2015 [W-29]; Vickery et al. 2017 [W-98]). Within the context of a high winds PRA, wind-driven rain considers rainwater deposition onto the equipment from vertical and horizontal exposure to rain, drip, and spray. Wind-driven rain does not include local flooding from intense rain. Flooding from rain or storm surge is considered a correlated hazard and is treated in other parts of this Standard. Wind-driven rain is therefore only considered at wind speeds above V_L, the lower bound starting wind speed for a high winds PRA. Rain effects for all other conditions are not part of a high winds PRA scope and are treated elsewhere in this Standard, such as External Flooding PRA. Several high winds PRAs considered wind-driven rain in a second phase of work. During the first phase, it was assumed that when the building envelope failed, the vulnerable electrical equipment was conditionally failed due to the potential for rainwater damage to the equipment. Subsequent refined analysis, with explicit modeling of envelope failures, wind direction frequency analysis, rain trajectory analysis, and equipment fragility development, produced less conservative fragilities. For example, a detailed wind-driven rain model (Vickery et al. 2017) showed that the electrical equipment fragilities from wind-driven rain were significantly lower than the fragilities of the enclosing building envelope, especially when directional wind and rain modeling was included in the analysis. There is a significant literature on wind-driven rain regarding building science (e.g., Blocken and Carmeliet, 2004 [W-99]). Information on rainfall rates, total rainfall, storm type, peak gust wind speeds, etc., can be obtained from the National Center for Environmental Information. Other useful references for modeling wind-driven rain effects include Blanchard and Spencer (1970) [W-100], Blevins (1984) [W-101], Dingle and Lee (1972) [W-102], and Willis and Tattleman (1989) [W-103]. In several high winds PRAs, the wind-driven rain analysis was coupled with 3-D progressive failure building models and directional wind analysis (Vickery et al. 2017) [W-98]. It is important to note that rain does not always accompany high wind events and the rain probability depends on the wind hazard type. Twisdale et al. (2015a) [W-76] noted that the probability of rain for thunderstorm and extratropical winds (within 24 hours of a high wind event) was in the range of 0.4 to 0.6 for several sites analyzed in North America. In addition,

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-73 (Cont'd)	wind-driven rain is typically a concern only for electrical equipment that is vulnerable to water deposition and leakage into the interior of the equipment (e.g., slow leakage through openings as a result of air pressure differences). Motor control centers housed in turbine or other metal clad buildings have been analyzed at several plants where wind-driven rain analysis was undertaken. NUREG/CR-6533, SAND97-1735 (Murata et al. 1977) [W-8] provide potentially useful vulnerability categories of electrical equipment, sorted according to steam-induced failures. See WFR-G1
W-N-74	Building envelope failures that permit entry of rainwater include failure of the roof cover, roof deck, doors, windows, vents, cladding, and other elements that enclose the building and protect SSCs vulnerable to wind-driven rain. See WFR-G2
W-N-75	Aleatory and epistemic uncertainties in fragility development and/or fragility curves are discussed and illustrated in numerous papers, including Banik et al. (2017) [W-19], Sciaudone et al. (2015 [W-13], 2017 [W-87]), Nicholas et al. (2017) [W-104], Hess et al. (2017) [W-105], Twisdale et al. (2015) [W-29], and Twisdale (2016 [W-35], 2017 [W-31]). See WFR-H1
W-N-76	Key assumptions and uncertainties regarding high wind PRAs are discussed in many of the above papers. Lovelace et al. (2017) [W-30] provide a useful summary. See WFR-H3, WFR-H4
W-N-77	For example, using uncertainty analysis or sensitivity studies. See WFR-H4
W-N-78	The purpose of this WPR-A1 is to ensure that proper consideration is given to the type of high wind event being considered. Tornadic events will have the ability to lift "missiles" in the wind stream and transport them to elevations much higher than their initial position. Tornadic winds also impose unique motions on the debris and may limit debris speed relative to the peak tornado speed. Straight wind events typically cover large areas and can produce missiles over wide areas. Straight wind motions may also allow increased acceleration of missiles over longer distances due to the absence of high rotational winds. The high winds PRA includes consideration of high wind events when the plant is initially at-power. However, the initiating high wind may result in actions to re-configure or change the operating mode of the plant prior to the onset of the high wind-induced initiating event or in response to warnings. Therefore, in addition to initiating events caused directly by the high wind event (e.g., LOOP, loss of ultimate heat sink availability, loss of functions due to loss of SSCs), this SR also requires that "indirect" initiating events be considered (e.g., initiating events caused by actions to shut down the plant or isolate the plant from the grid). This WPR-A1 also recognizes that human actions associated with plant shutdown or other plant high wind response activities may lead to initiating events. It is noted that the failure of certain SSCs may lead to multiple induced initiating events. See WPR-A1
W-N-79	Examples of industry experience may include a thorough review of reactor response to past high wind events or warnings, industry operating experience, as well as other available high wind risk evaluations for nuclear plants. See WPR-A3
W-N-80	Given the unique challenges that may arise during a high wind event, this requirement is intended to ensure that the analyst considers a range of available information sources related to high wind-related challenges to the plant. Relevant high wind experience may include events experienced at the site as well as industry operating experience. In addition, this SR requires reviewing situations where actions were taken in response to warnings. This SR requires that these experiences as well as other available high wind risk evaluations be reviewed as part of the development of the high winds PRA. See WPR-A3

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-81	The internal events PRA model may or may not include event sequences involving two or more reactors or radionuclide sources. However, if wind-induced failures impact two or more reactors or sources, such event sequences may be introduced for treatment of this hazard group. See WPR-B1, WPR-C1
W-N-82	The peer review for the PRA used as the base model could include findings that may not yet be resolved. This SR requires that the analyst(s) verify that findings from the peer review of the internal events PRA, which are relevant to the High Winds PRA, be addressed. Findings related to PRAs that are irrelevant to a high winds PRA (e.g., anticipated transient without scram, certain RCB breach sequences) need not be addressed as part of WPR-B2. This WPR-B2 also requires that the analysts ensure that the disposition of peer review findings does not adversely affect the development of the high wind plant response model. In some cases, the disposition of peer review findings from the internal events PRA may lead to an update of the high wind PRA plant response model. See WPR-B2
W-N-83	WPR-B3 addresses SSC functional failure modes, which are the result of the high wind impact upon the SSC (e.g., due to overtopping, crimping). This could result in SSC functional failure modes that were not identified in the internal events PRA. See WPR-B3
W-N-84	See commentary for WFR-A8 See WPR-B4
W-N-85	WPR-B5 addresses failures that are beneficial to the system. An example of a beneficial failure is the assumed failure of off-site power when a component would fail in the desired position. In general, beneficial failures are not typically included in the PRA unless exclusion of the failure would distort results. See WPR-B5
W-N-86	Mission time should consider industry experience in restoring off-site power following various high wind intensities. These times should be represented in reliability estimates of coping equipment. Coping times greater than 24 hours are possible particularly for long duration events such as hurricanes and extratropical cyclones. Use of convolution in power recovery may be considered to the extent supported by data. CC-I applications may include a single bounding mission time for all high wind scenarios. CC-II mission times may depend on high wind intensity. Where realistic times to power recovery may be less than the internal events mission time, the internal events mission times should be used. See WPR-B6
W-N-87	WPR-B9 ensures that multi-reactor effects are addressed within the PRA. For example, this SR is intended to (a) ensure resources credited to the reactor under analysis would be available given that other reactor(s) might compete for the same resource, (b) ensure the high winds PRA for one reactor captures the effect of failures at the other reactor(s) (e.g., failures of shared SSCs), and (c) ensure that the effect of on-site accessibility is captured. See WPR-B9
W-N-88	This SR is not applicable to PRAs performed for a single reactor plant. See WPR-B9
W-N-89	The internal event PRA model is the starting point for developing the HWEL. However, many SSCs that would be pertinent to high wind events may not have been identified in the internal events PRA. Consideration is required for barriers, doors, off-site power lines, tanks, and other equipment uniquely related to high wind response. An example approach can be seen in the EPRI high wind walkdown guidance document: (EPRI 3002008092, High Wind Equipment List and Walkdown Guidance [W-15]). See WPR-C1
W-N-90	It is expected that utilities with multiple site PRAs will regard that information as available for review. See WPR-C3

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-91	This SR addresses SSC functional failure modes (e.g., failure to open a valve) so as to confirm that the associated fragilities encompass all the SSC supporting components (e.g., instrument air piping). See WPR-C5
W-N-92	Operator actions evaluated for high winds need to take into account the unique timing and damage aspects of each high wind hazard (e.g., straight winds, tornadoes, and hurricanes). For example, tornadoes may occur with very little warning and only impact the plant for a short duration and hurricanes may come with substantial advanced warning but may impact the plant for a much longer duration. See WPR-D2
W-N-93	The intent of this requirement is to ensure that the interpretation of procedures as well as associated challenges is realistic. Because it is not possible to reasonably simulate many actions under the conditions that they will actually be performed (e.g., actions performed under high winds), judgment may be required when assessing manual actions. WPR-D5 is intended to strengthen the validity of the assessment by consulting with operators, personnel with knowledge of operations, or other personnel that may be performing actions. For example, response times for human actions taken under high wind conditions may increase relative to actions taken under nominal conditions. Moreover, delays in initiation of actions may result in the actions being taken instead under high wind conditions rather than as originally planned under nominal conditions or in delaying the initiation of these actions until conditions allow for the action to be performed safety (including transit to the plant location where the action is to be performed). See WPR-D5, WPR-D6
W-N-94	This SR is likely not applicable to PRAs performed during the pre-operational stage. See WPR-D5
W-N-95	This SR is likely not applicable to PRAs performed during the pre-operational stage as plant-specific procedures and trained plant staff may not be available. See WPR-D6
W-N-96	If pre-event actions are credited, ensure that adequate warning time is available and the environment where the action is being performed (e.g., in the control room versus in the yard) are appropriately considered. For actions taken during a high wind event, consider the potential for event related stresses. Post-event actions should consider the impact of debris and the potential for exterior doorways to be jammed. See WPR-D11
W-N-97	The effect of high wind hazard on the control room and ex-control room human actions include, for example, *(a)* additional workload and stress; *(b)* environment under which personnel are working (e.g., weather, heat, lighting, radiation); *(c)* high wind failures that impact access; *(d)* staffing and communications; *(e)* lack of cue availability; *(f)* effects of high wind on mitigation, required response, timing, accessibility, and potential for physical harm; *(g)* wind-specific job aids and training. See WPR-D11

Table W-1 Notes Supporting High Winds PRA Requirements (Cont'd)

Number	Notes
W-N-98	The PRA systems and event sequence model for a high wind PRA is commonly based on the internal events, at-power PRA systems model, to which a number of items are added such as high wind initiating events, SSC failure probability basic events derived from the fragility analysis, as well other basic events (e.g., new or adjusted HEPs for the specific hazard). Other factors to be considered include unique aspects of common causes, fragility correlations, any warning time available to take mitigating steps, and the possibility of recovery actions. Internal events event sequence models may also be modified, or some sequences not used for a given hazard model. Screening out certain parts of the internal events systems model from explicit incorporation in a hazard PRA model is common (the screening can take the form of explicitly deleting the logic in the hazard PRA or by bypassing or directly failing the logic, as appropriate). New system fault tree logic and/or event sequence logic may need to be developed and added into the PRA model. See WPR-E1
W-N-99	Certain quantification tools utilize approximations that may cause results to become inaccurate when success branches include basic events with high failure probabilities. In recognition of the possibility of high failure probabilities in conjunction with high wind specific actions or SSCs subjected to high wind conditions, this SR is intended to ensure the analyst(s) considers and addresses the limitations of computational tools when performing quantification. See WPR-E2
W-N-100	During calculation of the wind speed hazard points to be utilized in the quantification, the points should be discretized appropriately around the plant's unique vulnerabilities so as to allow for convergence of the event sequence family frequency. The analyst may demonstrate convergence on the risk metrics by performing sensitivity studies to show that event sequence family frequency does not significantly change with increased discretization. The analyst may choose and justify an alternate convergence level based on high wind impacts and total event sequence family frequency. See WPR-E3
W-N-101	This requires that the analyst perform appropriate assessments to confirm the correctness of the calculation process. For Event Sequence Quantification elements not explicitly considered, the analyst should document the basis for exclusion. See WPR-E4
W-N-102	Examples of items to be documented include (but are not limited to) the following: *(a)* key findings from investigations; *(b)* insights from operator interviews, talk-through(s), table-top exercises, or simulation(s), as available, to the extent the actions are credited in the high wind PRAs and would be impacted by the presence of high winds PRA phenomena; *(c)* high wind event trees and fault trees; *(d)* the specific adaptations made in the internal events PRA model to produce the high winds PRA model and the basis for those adaptations, or a description of ad-hoc models developed specifically for the high winds PRA; *(e)* the basis for selection of SSCs included in the high winds PRA, and associated fragilities of those SSCs; *(f)* the specific high wind-related influences that affect methods, processes, or assumptions used as well as the identification and quantification of the HFEs; *(g)* the recovery human actions included in the plant response model; *(h)* the preparatory human actions included in the plant response model; *(i)* significant risk contributors in the High Winds PRA model. See WPR-F1

W.2 EXPLANATORY MATERIAL ASSOCIATED WITH HIGH WINDS PRA

The non-LWR PRA Standard is anticipated to be used by non-LWR reactor designers and vendors prior to site selection (i.e., at the time of design certification application). The High Winds PRA, as well as external flooding, seismic, and other external hazard technical elements and sub-elements, will require a site-specific analysis (e.g., probabilistic wind hazard analysis). Prior to site selection, designers may seek to perform a high winds analysis of their design. The following guidelines are provided to aid designers and vendors in identifying a bounding set of analyses for high winds PRA and external hazards.

(a) The designer or vendor will select the design-basis external hazards (seismic, tornado, etc.) that generally envelop the potential sites where the plant would be located. For seismic events, seismic response analysis is done for different site conditions to obtain a bounding set of responses.

(b) At the time of the design certification, the fragilities of some SSCs may need to be estimated using generic information and design criteria. The high winds hazard (in terms of hazard curve and ground response spectra) is chosen to envelop the potential sites. The goal of a high winds PRA at the design certification application stage is to identify vulnerabilities and risk insights associated with the design. The PRA at this stage also assures the designer that the plant would meet the associated risk criteria when complete. Even without site-specific information, the high winds PRA should reveal any unique high wind-induced event sequences and event sequence families that could be efficiently addressed during the design. If the high wind-induced event sequences do not fit into existing release categories, new release categories are defined for which new mechanistic source terms and radiological consequences are needed. This will facilitate the inclusion of high wind-induced event sequences into the Risk Integration.

(c) After the design-certification application, a site “A” is chosen and the detailed design of the SSCs completed (or checked) using site-specific information. The hazard curve for the site is used in the quantification along with the plant and design-specific fragilities of SSCs. The high wind-induced risk is re-evaluated to capture the introduction of a site-specific hazard analysis and site and design dependent fragility analysis, and Risk Integration is accomplished using the same process as was used for the design certification.

(d) When a second site “B” is selected, the designer is expected to verify that the site “A” chosen after the design certification application is suitable at site “B”; any needed modification resulting from site characteristics at B will have to be made. Similarly, the high winds PRA will be modified to represent the site-specific conditions at “B” and the hazard curve for site “B.” The SRs that address site-specific information and conditions when a site has been selected should be applied.

Advanced non-LWRs are generally assumed to be simpler, have fewer systems, and rely more on inherent safety features to perform safety functions passively. In the external hazards elements of the PRA, this simplicity is represented in a simpler internal events PRA model that would produce a much shorter list of components for a fragility evaluation. However, the technical approach to hazards and fragility analysis are technology inclusive and hence, those requirements are believed to be comparable for a constructed or operating plant at a specific site. For plant designs that are simpler, the complexity of the external hazards PRA is expected to decrease.

W.2.1 Wind Hazard Analysis

For purposes of this Standard, the high wind hazard group includes the following wind hazard types:

(a) straight winds (thunderstorm winds and extratropical cyclones);

(b) tropical cyclone winds (hurricanes and typhoons); and

(c) tornadoes.

The specific definitions and sub-groupings of these high wind hazard types are an integral part of the scope, organization, and requirements. These distinct wind hazard types have separate phenomenological characteristics with observational data typically contained in separate databases, the analysis of which requires different methods of analysis. The types of wind hazards identified in Wind Hazard Analysis are consistent with the modern characterization of the phenomena and methods used for analysis for windstorms that have the capability to produce high wind speeds. The analyst should be familiar with the sources of the data and the uses and limitations of these databases. There is a significant literature regarding data sources and analysis methods for straight winds, tropical cyclones, and tornadoes.

Straight winds include thunderstorm winds and extratropical cyclones. High wind speeds in thunderstorms are associated with gust fronts, derechos, and downbursts. Extratropical storms are often referred to as winter storms, mid-latitude cyclones, or “Nor’easters” (in the eastern U.S.). These storms cover large areas, may have durations of hours to days, and may produce wind-driven storm surge for sites near large bodies of water.

Tropical cyclones have a low-pressure center and generally form over warm ocean water, predominately in the tropics. Tropical cyclones are often referred to as hurricanes, cyclones, or typhoons, depending on location and storm intensity. Tropical cyclones cover large areas, and the duration of high winds at a site can last for hours. Tropical cyclones are often accompanied by intense rain and may produce wind-driven storm surge.

A tornado is defined as “a rotating column of air, in contact with the surface, pendant from a cumuliform cloud, and often visible as a funnel cloud and/or circulating debris/dust at the ground” (AMS, 2013) [W-52]. It is important to note that the tornado literature emphasizes that tornado reporting is not efficient in low population areas and many tornadoes may go unreported in areas where few people live. Tornadoes occur with a wide range of intensities, lengths, and widths and may have single or multiple vortices. Tornado intensities are estimated from observed damage and, hence, the intensity levels (F and EF scales; Fujita, 1971 [W-28]; and TTU, 2006 [W-106]) must be converted to wind speeds in the tornado hazard analysis. Typically, there are significant uncertainties in estimating tornado hazard wind speed frequencies.

It is important to note that various national wind loading standards may refer to additional types or sub-types of wind hazards. For example, ASCE 7 [W-1], [W-2], and [W-3] refers to "special wind regions," which include mountainous terrain, gorges, and other complex terrain regions identified in the ASCE 7 wind speed maps. Sites in these areas may be subject to unusual wind conditions and may have high local wind speeds resulting from complex terrain and/or simple isolated topographic speedups (of the type included in ASCE 7). In addition, sites in arid or semi-arid locations may be subject to windstorms with significant amounts of entrained dust or sand. A separate hazard distinction for "special or unusual" wind conditions is not included in this technical element. The consensus is that plants sited in such locations would be very unusual so that the Wind Hazard Analysis requirements for site and regional anemometer analysis under the straight wind hazards would be sufficient to identify any "special wind" conditions that may exist at a site. In this regard, it is noted that topographic speed ups are included as a SR. Due to the highly site-specific nature of unusual wind environments, and following ASCE 7 recommendations for such locations, consultation with a wind engineer or meteorologist is advised if the site is deemed to be subject to "special winds." These analyses may require expert consultants, review of historical storm documentation and records, modeling, and/or wind tunnel testing to develop the information necessary for the hazard and fragility analyses.

Wind Hazard Analysis uses wind speed as the high winds independent hazard parameter, consistent with national and international codes and standards. Since wind pressure loads on rigid structures and components are proportional to the square of the wind speed, the use of wind speed as the independent wind hazard parameter introduces an inherent sensitivity to uncertainties in the wind speed frequencies. Wind-generated missile effects are generally proportional to a higher exponential power of wind speed due to the number of missiles produced and higher missile speeds that result from an increase in wind speed. Flexible structures are also proportional to the wind speed at a higher exponential power. Small changes or uncertainties in wind speed for a given return period can result in significant changes in analyzed load effects and in the failure frequency of a vulnerable SSC. Therefore, the development of mean frequencies, considering aleatory and epistemic uncertainties, is a critical part of the Wind Hazard Analysis and is essential to producing accurate high winds PRA results.

Site-specific wind hazard analysis generally requires the consideration of regional data. The size of the region requires judgment and depends on the regional climatology and type of wind hazard, the number of years in which accurate records are available, the extent and quality of the data, and the hazard's spatial variability within the region.

Care must be taken in understanding the sources and quality of the data in the site wind hazard analyses. For example, a "site" anemometer may be poorly sited, it may not include archived peak gust data, it may not have continuous records, or may not have sufficiently long records. There are no ready fixes for data produced from poorly sited anemometers. Differences in anemometer types, siting history, and conditions can produce a notable impact on the recorded wind speeds. The emphasis on data quality and analysis in Wind Hazard Analysis follows directly from widely recognized requirements in wind hazard modeling. For example, ASCE 7-16 (Section 26.5.3) [W-3] includes a list of requirements regarding analysis procedures when Wind Hazard Analysis is undertaken in lieu of the basic wind speeds provided in that standard. Understanding these and other limitations of wind hazard data is essential to producing wind hazard frequencies that are accurate and incorporate the appropriate uncertainties.

W.2.2 Wind Fragility Analysis

The Wind Fragility Analysis requires a systematic evaluation of the effects of wind on SSCs. The independent fragility variable is the reference wind speed defined in Wind Hazard Analysis. If the reference wind speed parameters are different for different wind hazard types, the fragility analysis must also incorporate this distinction.

Wind Fragility Analysis identifies four wind effects: (a) wind pressure and APC; (b) wind-generated missiles; (c) structural interactions; and (d) wind-driven rain. Wind pressure and APC effects are two separate effects that have been combined for purposes of this Standard since both may affect the net pressure loads on an SSC. An analyst may choose to develop separate fragilities for wind pressure and APC effects and indicate how they are combined. Wind-driven rain includes the effects of rain that may accompany high wind events. High wind events can produce failure of building envelopes (wall cladding, roof cover/deck, and openings), which provide pathways for rain to enter the structure and which may result in failure of electrical equipment vulnerable to vertical and horizontal rain and associated drips and sprays. Wind-driven rain does not include local flooding from a high wind event. If the rain in a high wind event can produce local internal or external flooding, that effect is considered as a correlated hazard.

Wind fragilities may be hazard-dependent. Hazard dependence means that the fragility depends on the characteristics of the particular high wind hazard type. That is, the SSC's fragility values for different wind hazards may not be the same for each wind speed value. For example, recent work suggests that wind-generated missile fragilities may be different for tropical cyclones (hurricanes) than for tornadoes of the same or similar wind speeds (NUREG/CR-7004 (Simiu and Potra, 2011) [W-9]; Twisdale, 2016 [W-35]). The analyst must assess the potential for fragilities being dependent on the wind hazard type and develop fragilities accordingly.

Wind fragility analysis methods follow the national standards, where appropriate, and use much of the same terminology. New work in high winds PRA fragility analysis has been completed in the past few years, and this information is referenced where appropriate.

The analyst should not interpret the commentary on Wind Fragility Analysis as limiting flexibility in the conduct of the technical analyses or in the application of expert and engineering judgment. A broad range of methods and judgment are required to analyze wind effects and develop fragilities for the dominant failure modes.

W.2.3 Wind Plant Response Analysis

To address the Wind Plant Response Analysis requirements contained herein, the high winds PRA analysis team should possess an internal events, at-power Level 1 and either a Level 2 or LERF PRA, developed either before or concurrently with the High Winds PRA. It is assumed that

(a) the internal events PRA will be used as the basis for the high winds PRA systems analysis (if appropriate);

(b) ideally, the internal events at-power PRA (if used) should have been peer reviewed and confirmed to be in compliance with other technical elements of this Standard.

Systems analysis for high winds PRAs may include both adding high wind-related basic events to the internal-events systems model as well as “trimming” some aspects of that model that do not apply or may be screened out.

W.2.4 Other High-Wind Resources

The sections that follow provide resources that may be useful to the analysts performing various activities associated with the high winds PRA. It is emphasized that, due to the current and evolving nature of the state of practice and experience related to high wind PRA, there is a great deal of diversity in the approaches described in these resources. Moreover, many of the resources were developed for applications not related to nuclear power plant applications. For this reason, the resources below are provided for information and, individually, should not be interpreted as providing definitive or authoritative references for meeting the requirements of this Standard.

The goal of many requirements in this technical element is to ensure that analysts appropriately consider known characteristics, challenges, and issues associated with high wind hazards as well as SSCs and plant response. In many cases, requirements in this technical element have been written to afford significant flexibility in how these topics are addressed within the high winds PRA. It is also recognized that expert and engineering judgment as well as implicit treatment of certain issues may be needed to facilitate development of the high winds PRA. In exercising this judgment, one may use the resources provided below to gain useful insights regarding the state of practice. Note that this list is not offered as an exhaustive list. Inclusion of these references does not constitute their endorsement by this Standard.

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4.3.13 External Flooding PRA (XF)

This Section presents the technical requirements associated with External Flooding PRA.

The requirements in this Section are divided into the following technical subelements:

(a) External Flood Hazard Analysis (XFHA);
(b) External Flood Fragility Analysis (XFFR);
(c) External Flood Plant Response Analysis (XFPR).

4.3.13.1 Objectives and Technical Requirements for External Flood Hazard Analysis (XFHA)

The objectives of the External Flood Hazard Analysis ensure that

(a) flood hazards applicable to the site are identified;

(b) flood hazard severities are characterized;

(c) frequencies of flood hazard(s) are based on a site-specific flood probabilistic flood hazard analysis;

(d) inputs are based upon current data;

(e) uncertainties at each step of the External Flood Hazard Analysis are identified and characterized;

(f) data and findings of investigations are incorporated;

(g) the External Flood Hazard Analysis is documented so as to provide traceability of the work.

Table 4.3.13.1-1 High Level Requirements for External Flood Hazard Analysis

Designator	Requirement
HLR-XFHA-A	Flood hazards, including combinations of flood hazards, that are applicable to the site and relevant to the External Flooding PRA shall be identified.
HLR-XFHA-B	The severity of the flood hazard shall be characterized.
HLR-XFHA-C	The frequency of the flood hazard severity for hazards at the site that did not screen out from further evaluation shall be based on a site-specific probabilistic flood hazard analysis that represents the state-of-knowledge and available data, models, and methods.
HLR-XFHA-D	Inputs to the External Flood Hazard Analysis shall be based on current data and information.
HLR-XFHA-E	Aleatory and epistemic uncertainties in each step of the External Flood Hazard Analysis shall be identified, characterized, and included in the final quantification of hazard estimates for the site.
HLR-XFHA-F	The External Flood Hazard Analysis shall incorporate the data and findings of an investigation(s) to establish or confirm either as-built, as-operated or as-designed, as-intended-to-operate conditions.
HLR-XFHA-G	Documentation of the External Flood Hazard Analysis shall provide traceability of the work.

Table 4.3.13.1-2 Supporting Requirements for HLR-XFHA-A

Flood hazards, including combinations of flood hazards, that are applicable to the site and relevant to the External Flooding PRA shall be identified. (HLR-XFHA-A)

Index No. XFHA-A	Capability Category I	Capability Category II
XFHA-A1	For the External Flood Hazard Analysis, either: *(a)* IDENTIFY the site at which the reactor being analyzed is located; or *(b)* DESCRIBE a bounding site and JUSTIFY that the bounding site bounds the list of sites in the scope of the PRA. See Note XF-N-1	
XFHA-A2	COMPILE a list of flood hazards that are applicable to the site. See Note XF-N-2	
XFHA-A3	IDENTIFY mechanisms associated with coexistent hazards and ADDRESS the impacts of coexistent hazards as part of screening out a flooding mechanism such that coexistent hazards do not invalidate screening analysis assumptions regarding the ability of the plant to respond to the mechanism. See Note XF-N-3	
XFHA-A4	COLLECT current information for the site and region to be used for identification of relevant floods. See Note XF-N-5, XF-N-6, XF-N-7	

Table 4.3.13.1-2 Supporting Requirements for HLR-XFHA-A (Cont'd)

Flood hazards, including combinations of flood hazards, that are applicable to the site and relevant to the External Flooding PRA shall be identified. (HLR-XFHA-A)

Index No. XFHA-A	Capability Category I / Capability Category II
XFHA-A5	If screening out the local intense precipitation (LIP) flood hazard, ENSURE that all of the criteria of the selected approach (either deterministic or probabilistic) are met: *(a)* Deterministic approach: USE the screening criteria of SCR-3 in Table 1.10-1, subject to meeting all of the following additional criteria: Criterion A: The site flood protection system is primarily composed of permanent and passive flood protection features with limited reliance on robust active features; Criterion B: Demonstrably conservative deterministic analysis of the impacts of the LIP hazard mechanism does not result in a flood severity that exceeds the capability of flood protection; Criterion C: There is margin demonstrated between the flood hazard from the demonstrably conservative deterministic analysis and the occurrence of cliff-edge effects; or *(b)* Probabilistic approach: Using demonstrably conservative assessment (or a realistic assessment that meets all applicable requirements of this Section), USE the hazard screening criteria of either SCR-1 or SCR-2 in Table 1.10-1. See Note XF-N-9
XFHA-A6	If screening out flood hazards other than the LIP flood hazard, ENSURE that all of the criteria of the selected approach (either deterministic or probabilistic) are met for each screened out mechanism: *(a)* Deterministic approach: USE the screening criteria of SCR-3 in Table 1.10-1, subject to meeting all of the following additional criteria: Criterion A: Demonstrably conservative deterministic analysis of the flood hazard, including associated combinations of mechanisms, does not result in a flood elevation that exceeds either the lowest of (i) the power block elevation or (ii) the elevation of flood response structures, systems, and components (SSCs) that could be exposed to floodwaters or water ingress. If an alternate elevation is used, JUSTIFY that there is minimal potential for the flood to affect or impinge upon flood response SSCs as well as credited flood protection features. Criterion B: There is margin between the conservatively defined flood hazard and the occurrence of cliff-edge effects; or *(b)* Probabilistic approach: Using demonstrably conservative assessment (or a realistic assessment that meets all applicable requirements of this Section), USE the hazard screening criteria of SCR-1 or SCR-2 in Table 1.10-1. See Note XF-N-10, XF-N-11
XFHA-A7	Using demonstrably conservative assessment (or a realistic assessment that meets all applicable requirements of this Section), ENSURE the event sequence family frequencies of external flood hazards probabilistically screened out from the external flood hazard group do not exceed the screening criteria SCR-2 in Table 1.10-1 or JUSTIFY use of alternate criteria. See Note XF-N-12, XF-N-13
XFHA-A8	CONFIRM that the external flood hazard screening represents either the as-built, as-operated or as-designed, as-intended-to-operate configuration of the plant, including relevant deficiencies (if applicable), and by performing investigations. See Note XF-N-14, XF-N-15
XFHA-A9	IDENTIFY the final set of flood hazards, or combinations of flood hazards, that did not screen out. See Note XF-N-16

Table 4.3.13.1-3 Supporting Requirements for HLR-XFHA-B

The severity of the flood hazard shall be characterized. (HLR-XFHA-B)

Index No. XFHA-B	Capability Category I	Capability Category II
XFHA-B1	CHARACTERIZE the site flood hazard using measures such as flood elevation, parameters related to associated effects, and flood event duration, including warning time for preparatory actions (e.g., implementation of temporary barriers) and period of inundation and ENSURE that the flood hazard characterization is consistent with the parameter(s) needed for subsequent fragility and plant response analysis. See Note XF-N-17, XF-N-18	
XFHA-B2	ESTIMATE the severity of flood hazards (see Requirement XFHA-B1) at different locations on the site by considering the effect of site features (e.g., topography and buildings). See Note XF-N-19, XF-N-20, XF-N-21	
XFHA-B3	For PRAs performed during the pre-operational stage, IDENTIFY assumptions related to the proposed changes to the topography and assumed placement of buildings made to satisfy Requirement XFHA-B1. See Note XF-N-21, XF-N-22	
XFHA-B4	GROUP hazards only when the following can be ensured: *(a)* the measures of flood severity (e.g., flood height, associated effects, and flood event duration) are similar; and *(b)* plant response strategies and characteristics are similar (e.g., use of similar protective and mitigating measures, similar operator actions, similar timing). If hazards are grouped into bins to facilitate analysis, SPECIFY the basis for the binning. See Note XF-N-23	

Table 4.3.13.1-4 Supporting Requirements for HLR-XFHA-C

The frequency of the flood hazard severity for hazards at the site that did not screen out from further evaluation shall be based on a site-specific probabilistic flood hazard analysis that represents the state-of-knowledge and available data, models, and methods. (HLR-XFHA-C)

Index No. XFHA-C	Capability Category I	Capability Category II
XFHA-C1	For each external flood hazard identified per Requirement XFHA-A9 or each group of hazards identified in XFHA-B4, IDENTIFY modeling methods and assumptions used to perform the probabilistic flood hazard analysis.	
XFHA-C2	JUSTIFY that the selections made in accordance with Requirement XFHA-C1 are demonstrably conservative. See Note XF-N-24	JUSTIFY that the selections made in accordance with Requirement XFHA-C1 are realistic to the extent practical. See Note XF-N-24
XFHA-C3	DEVELOP frequency estimates on flood hazard parameters (e.g., flood height or discharge; see XFHA-B1) using statistical methods or other probabilistic approaches. JUSTIFY that the methods and processes used are appropriate for the annual exceedance frequency range needed for the external flooding PRA (e.g., considering the quantity and quality of data used to select distribution functional forms and estimate parameters, appropriateness of models to characterize extrema). See Note XF-N-25	DEVELOP frequency estimates of flood hazard parameters (e.g., flood height or discharge; see XFHA-B1) using statistical methods or other probabilistic approaches. JUSTIFY that the methods and processes used are appropriate for the annual exceedance frequency range needed for the external flooding PRA (e.g., considering the quantity and quality of data used to select distribution functional forms and estimate parameters, appropriateness of models to characterize extrema) and consider mechanistic and phenomenological characteristics of the hazard, to the extent supported by the state of practice. See Note XF-N-25

Table 4.3.13.1-4 Supporting Requirements for HLR-XFHA-C (Cont'd)

The frequency of the flood hazard severity for hazards at the site that did not screen out from further evaluation shall be based on a site-specific probabilistic flood hazard analysis that represents the state-of-knowledge and available data, models, and methods. (HLR-XFHA-C)

Index No. XFHA-C	Capability Category I	Capability Category II
XFHA-C4	JUSTIFY that the process used to extrapolate and interpret data is appropriate through consideration of phenomenological and mechanistic factors (e.g., hydrology, oceanography, geotechnical engineering, and meteorology; see Requirements XFHA-C6, XFHA-C7, XFHA-C8, XFHA-C9, and XFHA-C10 for potentially relevant mechanism-specific considerations) as well as statistical analyses. See Note XF-N-26	
XFHA-C5	In developing the probabilistic flood hazard analysis results for use in Event Sequence Quantification, EXTEND the flood severity range to sufficiently severe values (consistent with the physical data and interpretations) so that the truncation does not distort the quantitative results (e.g., on parameters such as event sequence family frequencies), the associated risk insights, and the delineation and ranking of flood-induced sequences are not affected. See Note XF-N-27	
XFHA-C6	In developing a LIP hazard analysis, ADDRESS the following factors or DETERMINE that the factor is not applicable to the analysis or not necessary (e.g., hazard results are not sensitive to the factor): *(a)* local and regional precipitation history and basis for selecting applicable local and regional data; *(b)* aleatory variability resulting from variations in storm loading patterns (e.g., precipitation depth, area, and duration, as well as spatial and temporal distribution of precipitation); *(c)* site, building, and flood barrier layouts as well as drainage and run-off capabilities (including site and roof drainage); *(d)* limitations of data collection instruments relied on for data used to estimate frequencies; *(e)* potential impact of on-site impoundments (e.g., levees and dams); *(f)* local watershed effects on run-off; *(g)* uncertainties associated with hydrologic and hydraulic models. SPECIFY the basis for selection or treatment of the following: *(a)* storms (e.g., historical events) included in the storm catalog, if stochastic sampling or transposition is used; *(b)* time intervals, spatial resolutions, and seasonal/annual rainfall; *(c)* initial and boundary conditions as well as assumptions in losses and run-off conditions in modeling. See Note XF-N-28, XF-N-29	

Table 4.3.13.1-4 Supporting Requirements for HLR-XFHA-C (Cont'd)

The frequency of the flood hazard severity for hazards at the site that did not screen out from further evaluation shall be based on a site-specific probabilistic flood hazard analysis that represents the state-of-knowledge and available data, models, and methods. (HLR-XFHA-C)

Index No. XFHA-C	Capability Category I / Capability Category II
XFHA-C7	In developing a riverine flooding hazard analysis, ADDRESS the following or DETERMINE that the factor is not applicable to the analysis or not necessary (e.g., hazard results are not sensitive to the factor): *(a)* flood history of the watershed including local and regional precipitation and river levels; *(b)* variations in storm loading patterns (e.g., precipitation depth, area, and duration as well as spatial and temporal distribution); *(c)* warm and cold season events (e.g., conditions associated with melting of winter snowpack); *(d)* limitations of data collection instruments relied on for data used to estimate frequencies; *(e)* engineered and natural features (e.g., permanent and temporary site features and downstream impoundments) affecting flood severity; *(f)* uncertainties associated with hydrologic and hydraulic models; *(g)* wind-wave run-up. JUSTIFY for the site or ranges of sites the selection or treatment of the following: *(a)* watershed initial and boundary conditions; *(b)* watershed hydrologic and hydraulic models; *(c)* parameters describing watershed characteristics (e.g., roughness, channel profiles, channel cross-sections, ineffective flow areas). See Note XF-N-29, XF-N-30
XFHA-C8	In developing a hazard analysis for riverine flooding with upstream and downstream dams or flooding caused by on-site impoundments (if applicable), ADDRESS the following or DETERMINE that the factor is not applicable to the analysis or not necessary (e.g., consideration of the factor would not affect risk insights): *(a)* the factors listed in Requirement XFHA-C7, considering changes as a result of upstream dams; *(b)* relevant failure mechanisms (e.g., seismic, hydrological, and other failures of on-site and off-site impoundments or dams); *(c)* dam operational history (releases, reservoir levels, and reservoir capacities); *(d)* dam/impoundment condition (e.g., from condition/inspection reports or other assessments), age, and refurbishments; *(e)* type of dam/impoundment (e.g., dam construction and relevant design characteristics); *(f)* effects of dam operations (e.g., operating guidelines and controlled releases); *(g)* treatment of upstream and downstream dam failure; *(h)* uncertainties associated with dam/impoundment breach parameter estimates and breach modeling; *(i)* representative dam/impoundment failure scenarios with associated uncertainties for each scenario. In cases where the above information cannot be obtained for particular dams, SPECIFY the basis for the approach used to analyze the dams in the absence of the information. JUSTIFY selection or treatment of the following: *(a)* the justification factors listed in Requirement XFHA-C7; *(b)* the approach used for estimation of dam/impoundment failure frequencies; *(c)* dams and/or other water impounding structures not included in the analysis (including the method used to distinguish between consequential and non-consequential dams). See Note XF-N-29, XF-N-31

Table 4.3.13.1-4 Supporting Requirements for HLR-XFHA-C (Cont'd)

The frequency of the flood hazard severity for hazards at the site that did not screen out from further evaluation shall be based on a site-specific probabilistic flood hazard analysis that represents the state-of-knowledge and available data, models, and methods. (HLR-XFHA-C)

Index No. XFHA-C	Capability Category I	Capability Category II
XFHA-C9	In developing a surge hazard analysis for pressure-induced seiche, extratropical storm, or hurricane hazard mechanisms (if applicable), ADDRESS the following or DETERMINE that the factor is not applicable to the analysis or not necessary (e.g., consideration of the factor would not affect risk insights): *(a)* contribution from tropical and extratropical events; *(b)* historical data augmented, as appropriate, by other (e.g., synthetic) data sources, filtered for the geographic region, including identification of limitations of available data; *(c)* numerical wind field and surge models; *(d)* uncertainties associated with numerical models; *(e)* engineered and natural features (e.g., permanent and temporary site features) affecting flood severity. In developing a surge probabilistic flood hazard analysis for pressure-induced seiche, extratropical storm, or hurricane hazard mechanisms, JUSTIFY selection or treatment of the following: *(a)* methodology (e.g., Joint Probability Method or Empirical Simulation Technique); *(b)* filtering of data; *(c)* storm recurrence rate; *(d)* relevant storm parameters and their distributions; *(e)* application of wind and surge models; *(f)* bathymetry and topography; *(g)* uncertainties associated with the above. See Note XF-N-29, XF-N-32	
XFHA-C10	In developing a tsunami or seismic-induced seiche hazard analysis (if applicable), ADDRESS the following or DETERMINE that the factor is not applicable to the analysis or not necessary (e.g., hazard results are not sensitive to the factor): *(a)* tsunami or seiche hazard mechanisms (e.g., near- and far-field seismic faults, landslides or volcanoes, meteor-induced) and basis for hazard frequency; *(b)* current data used to develop bathysphere and topography for the site or bounding site, including anticipated site changes that may affect overland flow; *(c)* engineered and natural features (e.g., permanent and temporary site features) affecting flood severity. JUSTIFY the selection of tsunami or seiche hazard analysis methods and approaches, including source characterization; modeling approaches; and selection of hazard parameters (e.g., wave height, speed, and periodicity). See Note XF-N-29, XF-N-33	
XFHA-C11	For each external flood hazard identified per Requirement XFHA-A9, IDENTIFY coexistent hazards (if applicable) to support assessments required by the fragility and plant response. See Note XF-N-29, XF-N-34	

Table 4.3.13.1-5 Supporting Requirements for HLR-XFHA-D

Inputs to the External Flood Hazard Analysis shall be based on current data and information. (HLR-XFHA-D)

Index No. XFHA-D	Capability Category I	Capability Category II
XFHA-D1	For each flood hazard identified in Requirement XFHA-A9, including combinations of mechanisms, COMPILE current data and information for the sites for the relevant phenomena consistent with the models and methods selected. See Note XF-N-5, XF-N-35, XF-N-36	For each flood hazard identified in Requirement XFHA-A9, including combinations of mechanisms, COMPILE current data and information for the sites for the relevant phenomena consistent with the models and methods selected and AUGMENT with additional regional, global, and paleoflood data (as available and applicable). See Note XF-N-36
XFHA-D2	As applicable to the flood hazard considered, USE regional information and surveys to do the following: *(a)* develop regional topography and bathymetry; and *(b)* characterize land use and land cover.	
XFHA-D3	USE information collected to conservatively define parameters required for the probabilistic flood hazard analysis (e.g., meteorological, hydrologic, or hydraulic model parameters or initial conditions). See Note XF-N-37	USE information collected to realistically, as practical, define parameters required for the probabilistic flood hazard analysis (e.g., meteorological, hydrologic, or hydraulic model parameters or initial conditions). JUSTIFY that use of conservative model parameters, if used, is appropriate (e.g., does not distort risk insights). See Note XF-N-37
XFHA-D4	For regulated river systems, COLLECT relevant information regarding river operations (e.g., operating curves and procedures) and operating history. See Note XF-N-38	

Table 4.3.13.1-6 Supporting Requirements for HLR-XFHA-E

Aleatory and epistemic uncertainties in each step of the External Flood Hazard Analysis shall be identified, characterized, and included in the final quantification of hazard estimates for the site. (HLR-XFHA-E)

Index No. XFHA-E	Capability Category I	Capability Category II
XFHA-E1	IDENTIFY assumptions and sources of uncertainty in each step of the External Flood Hazard Analysis in a manner that supports Requirements XFHA-E2 and XFPR-F7.	
XFHA-E2	CHARACTERIZE notable sources of uncertainty for each flood hazard (e.g., using sensitivity studies related to alternate data, models, and methods). See Note XF-N-39	
XFHA-E3	ESTIMATE uncertainties in final quantification of the flood hazard using demonstrably conservative assumptions. ASSESS how conservative assumptions affect key insights and conclusions. See Note XF-N-39	PROPAGATE uncertainties in the final quantification of the flood hazard. See Note XF-N-39
XFHA-E4	ESTIMATE representative hazard frequency functions (often referred to as hazard curves) for each mechanism or group of mechanisms. JUSTIFY that the approach used yields a demonstrably conservative estimate of hazard frequencies and addresses notable sources of uncertainties. See XFHA-E2 See Note XF-N-40	QUANTIFY hazard frequency functions (often referred to as hazard curves), including mean and percentile hazard functions, for each mechanism or group of mechanisms identified in Requirement XFHA-A9. See Note XF-N-40

Table 4.3.13.1-7 Supporting Requirements for HLR-XFHA-F

The External Flood Hazard Analysis shall incorporate the data and findings of an investigation(s) to establish or confirm either as-built, as-operated or as-designed, as-intended-to-operate conditions. (HLR-XFHA-F)

Index No. XFHA-F	Capability Category I	Capability Category II
XFHA-F1	COLLECT information via investigation(s) (complemented, as needed, by hydrologic surveys) about either the as-built, as-operated or as-designed, as-intended-to-operate, as applicable, plant and site characteristics relevant to the hazard analysis such as site topography, features that may affect flow around the site, drainage features, and features that may impound water. See Note XF-N-15	
XFHA-F2	IDENTIFY pathways for water movement/ingression and potentially inundated areas of the plant and sites.	
XFHA-F3	IDENTIFY relevant deficiencies with respect to either as-built, as-operated or as-designed, as-intended-to-operate configuration of the plant that may affect the severity of flood hazards (e.g., degraded flood barriers or changes to site topographies that may alter site drainage).	
XFHA-F4	ADDRESS the information collected in Requirements XFHA-F1, XFHA-F2, and XFHA-F3 in the External Flood Hazard Analysis, External Flood Fragility Analysis, and External Flood Plant Response Analysis (as appropriate).	

Table 4.3.13.1-8 Supporting Requirements for HLR-XFHA-G

Documentation of the External Flood Hazard Analysis shall provide traceability of the work. (HLR-XFHA-G)

Index No. XFHA-G	Capability Category I	Capability Category II
XFHA-G1	DOCUMENT the process used in the External Flood Hazard Analysis specifying what is used as input, the applied methods and the results. Address the following, as well as other details needed to fully document how the set of Supporting Requirements (SRs) are satisfied: *(a)* the specific site or bounding site; *(b)* process used to identify and screen out external flood hazards; *(c)* the basis for screening out any external flood hazard or combination of flood hazards (including describing how coexistent hazards were addressed); *(d)* sources of uncertainty and related assumptions associated with the flood hazard analysis; *(e)* the following for flood hazards that were not screened out: *(1)* the approach used to perform the probabilistic flood hazard analysis; *(2)* the data, models, and methods used for determining the external flood hazard function; *(3)* the basis for including or excluding data, models, and methods in the analysis *(4)* assumptions and associated technical bases; *(5)* treatment of epistemic and aleatory uncertainties, including the approaches used for identifying, characterizing, and including uncertainties in the probabilistic flood hazard analysis. *(f)* the results of the External Flood Hazard Analysis.	
XFHA-G2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement XFHA-E1) associated with the External Flood Hazard Analysis.	
XFHA-G3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the External Flood Hazard Analysis. As identified in Requirements XFHA-B3 and XFHA-E1. See Note XF-N-21	
XFHA-G4	For External Flood Hazard Analysis PRAs using a bounding site, DOCUMENT the basis for the selection of the bounding site characteristics that bound the range of sites for which the plant is designed and the justification of the applicability of the bounding site. See Note XF-N-41	

4.3.13.2 Objectives and Technical Requirements for External Flood Fragility Analysis (XFFR)

The objectives of the External Flood Fragility Analysis ensure that

(a) fragilities of SSCs are identified whose failure may contribute to radioactive material release for each flood hazard;
(b) data and findings of investigation(s) of the plant focus on either as-built, as-operated or as-designed, as intended-to-operate conditions;
(c) fragility evaluations focus on flood characteristics that the SSCs experience at their location for each flood hazard;
(d) fragility evaluations are performed for relevant failure modes of SSCs;
(e) coexistent hazards performance on SSCs are assessed; and
(f) the External Flood Fragility Analysis is documented so as to provide traceability of the work.

Table 4.3.13.2-1 High Level Requirements for External Flood Fragility Analysis

Designator	Requirement
HLR-XFFR-A	The External Flood Fragility Analysis shall address, for each flood hazard, the fragilities of SSCs whose failure may contribute to the release of radionuclide material.
HLR-XFFR-B	The External Flood Fragility Analysis shall incorporate the data and findings of investigation(s) of the plant focusing on as-built, as-operated or as-designed, as-intended-to-operate site conditions.
HLR-XFFR-C	The External Flood Fragility Analysis shall be based on flood characteristics (e.g., flood height, associated effects, and flood event duration) that the SSCs experience at their location for each flood hazard.
HLR-XFFR-D	The External Flood Fragility Analysis shall be performed for relevant failure modes of SCCs affecting functions modeled in the plant response analysis.
HLR-XFFR-E	Effects of coexistent hazards on fragility shall be assessed.
HLR-XFFR-F	Documentation of the External Flood Fragility Analysis shall provide traceability of the work.

Table 4.3.13.2-2 Supporting Requirements for HLR-XFFR-A

The External Flood Fragility Analysis shall address, for each flood hazard, the fragilities of SSCs whose failure may contribute to the release of radionuclide material. (HLR-XFFR-A)

Index No. XFFR-A	Capability Category I	Capability Category II
XFFR-A1	IDENTIFY SSCs for which a fragility analysis is needed based on the external flood equipment list (XFEL) developed in accordance with Requirement XFPR-D1. INCLUDE in the scope of the External Flood Fragility Analysis those SSCs and associated functions identified from the plant response analysis (see the SRs of HLR-XFPR-D). See Note XF-N-42, XF-N-43	
XFFR-A2	ASSESS the impact of fluids which may be chemically or physically incompatible with water that are contained within SSCs in the XFEL on the External Flooding PRA plant response model (PRM). See Note XF-N-44	
XFFR-A3	If screening out SSCs from further evaluation is performed as part of the fragility analysis, ENSURE that at least one of the criteria of the selected approach (either deterministic or probabilistic) are met: *(a)* Deterministic approach: USE the screening criteria of SCR-3 in Table 1.10-1, subject to meeting all of the following additional criteria: Criterion A: It is not physically possible for the failure of SSC to impact plant systems as assessed by a demonstrably conservative, deterministic analysis; Criterion B: The failure of the SSC does not impact a mitigation function being considered in the PRA and the item does not result in (or create) another initiating event; or *(b)* Probabilistic approach: Using demonstrably conservative assessment (or a realistic assessment that meets all applicable requirements of this Section), USE the screening criteria of SCR-2 in Table 1.10-1. See Note XF-N-42, XF-N-45	

Table 4.3.13.2-2 Supporting Requirements for HLR-XFFR-A (Cont'd)

The External Flood Fragility Analysis shall address, for each flood hazard, the fragilities of SSCs whose failure may contribute to the release of radionuclide material. (HLR-XFFR-A)

Index No. XFFR-A	Capability Category I	Capability Category II
XFFR-A4	INCLUDE in the fragility assessment credited active and passive flood protection features, including permanent and temporary flood barriers (including seals) and equipment. See Note XF-N-42, XF-N-46	
XFFR-A5	If screening out SSCs, ADDRESS coexistent hazards in the screening assessment (see the SRs of HLR-XFFR-E) (i.e., coexistent hazards do not invalidate screening analysis assumptions regarding risk-significant contributions of SSCs). See Note XF-N-42, XF-N-47	

Table 4.3.13.2-3 Supporting Requirements for HLR-XFFR-B

The External Flood Fragility Analysis shall incorporate the data and findings of investigation(s) of the plant focusing on as-built, as-operated or as-designed, as-intended-to-operate site conditions. (HLR-XFFR-B)

Index No. XFFR-B	Capability Category I	Capability Category II
XFFR-B1	COLLECT information via investigation(s) about either the as-built, as-operated or as-designed, as-intended-to-operate, as applicable, plant and site characteristics relevant to the fragility evaluation, such as establishing or confirming the location and characteristics of flood protection features, penetrations/seals, and drainage features. See Note XF-N-15, XF-N-48	
XFFR-B2	For operating reactors, ASSESS the condition (e.g., SSC degradation) and configuration of SSCs observed during the investigation(s). See Note XF-N-15, XF-N-49, XF-N-50	
XFFR-B3	IDENTIFY water ingression paths, potentially inundated areas, and potentially impacted SSCs due to flood characteristics (e.g., height, associated effects, and flood event duration). See Note XF-N-51	
XFFR-B4	IDENTIFY potential flood-induced SSC interactions that may compromise the intended functions of SSCs or operator actions (e.g., interactions arising from buoyant SSCs impacting other SSCs or blocking pathways, external flood-induced failures leading to damage of other SSCs, chemical reactions with incompatible fluids). See Note XF-N-52	
XFFR-B5	ADDRESS the information collected in Requirements XFFR-B1, XFFR-B2, XFFR-B3, and XFFR-B4 in the External Flood Fragility Analysis and plant response analysis (as appropriate). See Note XF-N-42, XF-N-53	

Table 4.3.13.2-4 Supporting Requirements for HLR-XFFR-C

The External Flood Fragility Analysis shall be based on flood characteristics (e.g., flood height, associated effects, and flood event duration) that the SSCs experience at their location for each flood hazard. (HLR-XFFR-C)

Index No. XFFR-C	Capability Category I	Capability Category II
XFFR-C1	By using a conservative approach, IDENTIFY relevant failure mechanisms associated with the failure modes included in the plant response model. See Note XF-N-42, XF-N-54	For SSCs that are risk-significant to plant external flood response (or SSCs that are located in areas that are risk-significant contributors), IDENTIFY relevant realistic failure mechanisms (e.g., inundation, overturning, and structural failure) associated with the failure modes included in the plant response model. For SSCs that are not risk-significant to plant flood response and SSCs that are located in areas that are not risk-significant contributors, IDENTIFY relevant failure mechanisms associated with the failure modes included in the plant response model using a realistic or conservative approach. See Note XF-N-42, XF-N-54
XFFR-C2	For each SSC requiring a fragility assessment (see HLR-XFFR-A), ASSESS potential impacts from flood characteristics (e.g., flood height, associated effects, and flood event duration) associated with each flood hazard. See Note XF-N-42, XF-N-55	

Table 4.3.13.2-5 Supporting Requirements for HLR-XFFR-D

The External Flood Fragility Analysis shall be performed for relevant failure modes of SCCs affecting functions modeled in the plant response analysis. (HLR-XFFR-D)

Index No. XFFR-D	Capability Category I	Capability Category II
XFFR-D1	ESTIMATE representative fragilities for failure modes included in the plant response model. JUSTIFY that the fragilities are conservative and appropriate given the characteristics of the flood and SSC failure mechanisms (e.g., use of step functions for inundation-induced failures). See Note XF-N-42, XF-N-56	CALCULATE fragilities for failure modes included in the plant response model. JUSTIFY that the chosen form of each fragility is appropriate given the characteristics of the flood and SSC failure mechanisms (e.g., use of step functions for inundation-induced failures and fragility curves for static and dynamic effects, including debris loading). See Note XF-N-42, XF-N-56
XFFR-D2	In the development of SSC fragility, USE applicable generic information and technical evaluation. JUSTIFY that the use of generic information or technical evaluation is consistent with the demonstrably conservative nature of the Capability Category I (CC-I) assessment. See Note XF-N-42, XF-N-57	In the development of fragilities, USE either *(a)* plant-specific information, if available; or *(b)* generic information augmented by a plant-specific technical evaluation. JUSTIFY that the use of generic information or technical evaluations is necessary (e.g., plant-specific information is not available and technical evaluation is not practical) and yields fragilities that are appropriate for the site (e.g., account for key site-specific considerations). See Note XF-N-42, XF-N-57
XFFR-D3	IDENTIFY assumptions and sources of uncertainty in the fragility analysis in a manner that supports Requirements XFFR-D4 and XFPR-F7. See Note XF-N-42	
XFFR-D4	CHARACTERIZE notable sources of uncertainty in the fragility analysis (e.g., using uncertainty analysis or sensitivity studies).	

Table 4.3.13.2-6 Supporting Requirements for HLR-XFFR-E

Effects of coexistent hazards on fragility shall be assessed. (HLR-XFFR-E)

Index No. XFFR-E	Capability Category I	Capability Category II
XFFR-E1	IDENTIFY the SSCs on the XFEL that are susceptible to the effect of coexistent hazards based on review of characteristics of the external hazard mechanisms. See Note XF-N-58	
XFFR-E2	For each SSC not screened out in the SRs of HLR-XFFR-A, ADDRESS the effects of coexistent hazards on the performance of SSCs (i.e., whether coexistent hazards invalidate assumptions regarding the performance of SSCs under the flooding event). See Note XF-N-42, XF-N-59	

Table 4.3.13.2-7 Supporting Requirements for HLR-XFFR-F

Documentation of the External Flood Fragility Analysis shall provide traceability of the work. (HLR-XFFR-F)

Index No. XFFR-F	Capability Category I	Capability Category II
XFFR-F1	DOCUMENT the process used in the External Flood Fragility Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* screening methodology; *(b)* SSCs (including flood protection features and other SSCs that may be affected by the flood event) included in the fragility analysis; *(c)* locations (e.g., elevations) of SSCs; *(d)* the methodologies used to establish the flood-induced fragilities of SSCs; *(e)* the method of identifying SSC failure mechanisms, the identified failure mechanisms, and the associated failure modes; *(f)* the fragilities, parameter values, and the associated technical bases; *(g)* sources of information used in the fragility analysis; *(h)* investigation procedures; *(i)* investigation team composition and member qualification; *(j)* investigation observations and conclusions; *(k)* results of the fragility analysis; *(l)* sources of uncertainty associated with the external flood fragility evaluation.	
XFFR-F2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified Requirement XFFR-D3) associated with the External Flood Fragility Analysis.	
XFFR-F3	For PRAs performed during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated design or site details associated with the External Flood Fragility Analysis. As identified in Requirement XFFR-D3. See Note XF-N-21	

4.3.13.3 Objectives and Technical Requirements for External Flood Plant Response Analysis (XFPR)

The objectives of the plant response analysis ensure that

(a) external flood scenarios are developed for use with the plant response model;
(b) external flood-induced initiating events are included that cause risk-significant event sequences;
(c) external flood-induced SSC failures, non-external flood-induced SSC failures, unavailabilities, human errors, and multi-reactor effects are included that can lead to a release of radioactive materials;
(d) the list of SSCs for external flood fragility evaluations include SSCs that contribute to the External Flood Plant Response Analysis;
(e) mission times are defined based on the portion of the flood event duration for which the SSC operates;
(f) actions are included that are associated with the plant response strategies as well as actions that can be impacted by external flood-specific challenges;
(g) quantification is performed on event sequences by integrating external flood hazard, fragilities, and plant response, including uncertainties; and
(h) the External Flood Plant Response Analysis is documented so as to provide traceability of the work.

Table 4.3.13.3-1 High Level Requirements for External Flood Plant Response Analysis

Designator	Requirement
HLR-XFPR-A	The External Flood Plant Response Analysis shall systematically develop a collection of plant- or design-specific external flood scenarios for use in the external flood plant response model.
HLR-XFPR-B	The External Flood Plant Response Analysis shall include external flood-caused initiating events that cause risk-significant event sequences and/or risk-significant event progression sequences.
HLR-XFPR-C	The External Flood Plant Response Analysis shall include external flood-induced SSC failures, non- external flood-induced SSC failures, unavailabilities, human errors, and multi-reactor effects that lead to release of radionuclide material.
HLR-XFPR-D	The list of SSCs selected for External Flood Fragility Analysis shall include the SSCs that contribute to event sequences included in the External Flood Plant Response Analysis.
HLR-XFPR-E	The External Flood Plant Response Analysis shall include actions associated with plant response strategies (e.g., human actions to prepare for the flood event) and external flood-specific challenges to human performance.
HLR-XFPR-F	The analysis to quantify event sequence family frequencies shall integrate the external flood hazard, the external flood fragilities, and the external flood plant response, including uncertainties.
HLR-XFPR-G	The External Flood Plant Response Analysis shall incorporate the data and findings of an investigation(s) to establish or confirm as-built, as-operated or as-designed, as-intended-to-operate conditions.
HLR-XFPR-H	The documentation of the External Flood Plant Response Analysis shall provide traceability of the work.

Table 4.3.13.3-2 Supporting Requirements for HLR-XFPR-A

The External Flood Plant Response Analysis shall systematically develop a collection of plant- or design-specific external flood scenarios for use in the external flood plant response model. (HLR-XFPR-A)

Index No. XFPR-A	Capability Category I	Capability Category II
XFPR-A1	For each flood hazard listed per Requirement XFHA-A9, IDENTIFY the potential flood-related hazard scenarios that can cause a plant initiating event [e.g., loss of off-site power (LOOP), plant trip] and DESCRIBE key characteristics for each scenario, including the following: *(a)* flood scenario characteristics (e.g., flood elevation, associated affects, and flood event duration, including credited warning time); *(b)* credible flood protection failure mode(s), as applicable; *(c)* direct consequences of flood protection failure or consequences due to a lack of (sufficient) flood protection, as applicable; see Requirement XFPR-A5; *(d)* resultant plant impacts, including initiating events; *(e)* human actions associated with plant flood response; see Requirement XFPR-E1; *(f)* SSCs that can mitigate an initiating event as well as the features that protect those SSCs, if credited; *(g)* other relevant plant- or design-specific factors. See Note XF-N-60	
XFPR-A2	For scenarios leading to ingress of water into plant structures or areas containing credited SSCs, IDENTIFY *(a)* the potential sources of flooding, including pathways for ingress of water into the plant from external sources and from other internally flooded areas (e.g., through penetrations, through doors or other openings, or as a result of failure or overtopping of flood barriers); *(b)* the resultant flooded areas of the plant; and *(c)* the characteristics of water in flooded areas (e.g., volume, duration, and temperature), as needed for subsequent evaluations. See Note XF-N-61	
XFPR-A3	For multi-reactor sites with shared SSCs, IDENTIFY propagation pathways due to potential multi-reactor or cross-reactor impacts (e.g., potential impacts from flooding of an adjacent reactor). See Note XF-N-62, XF-N-63	
XFPR-A4	When choosing to screen out areas as not being susceptible to flooding, USE SCR-2 or SCR-3 in Table 1.10-1, or JUSTIFY alternate criteria. See Note XF-N-64	
XFPR-A5	For scenarios leading to ingress of water into plant structures or areas containing credited SSCs, DEVELOP external flood propagation scenarios by treating the external floodwater ingress as the flood source.	
XFPR-A6	DEFINE timelines for each scenario characterized in Requirement XFPR-A1. See Note XF-N-65	
XFPR-A7	If grouping scenarios, SATISFY the CC-I Requirements of FLEV-A1 and FLEV-A3, replacing internal flood considerations with external flood considerations.	If grouping scenarios, SATISFY the Capability Category II (CC-II) Requirements of FLEV-A1 and FLEV-A3, replacing internal flood considerations with external flood considerations.

Table 4.3.13.3-3 Supporting Requirements for HLR-XFPR-B

The External Flood Plant Response Analysis shall include external flood-caused initiating events that cause risk-significant event sequences and/or risk-significant event progression sequences. (HLR-XFPR-B)

Index No. XFPR-B	Capability Category I	Capability Category II
XFPR-B1	Using a systematic process and a review of relevant industry experience, IDENTIFY initiating events (e.g., due to failures of SSCs or human actions) caused directly or indirectly by the external flood event, including initiating events associated with changes in the plant mode or proceduralized plant reconfigurations (if applicable) due to the external flood event. INCLUDE additional event sequences, as applicable, associated with external flood-induced initiating events that are not included in the internal events PRA model such as events that impact two or more reactors or sources of radioactive material. See IE-A16 See Note XF-N-66, XF-N-67	
XFPR-B2	Using a systematic process and a review of relevant industry experience, IDENTIFY external flood-induced hazard events resulting from coexistent hazards that can induce initiating events or fail SSCs modeled in the external flooding PRA. See Note XF-N-66	
XFPR-B3	INCLUDE in the plant response model the initiating events, identified in Requirement XFPR-B1 and XFPR-B2, that cause risk-significant event sequences and/or risk-significant event progression sequences. See Note XF-N-68	

Table 4.3.13.3-4 Supporting Requirements for HLR-XFPR-C

The External Flood Plant Response Analysis shall include external flood-induced SSC failures, non- external flood-induced SSC failures, unavailabilities, human errors, and multi-reactor effects that lead to release of radionuclide material. (HLR-XFPR-C)

Index No. XFPR-C	Capability Category I	Capability Category II
XFPR-C1	ENSURE that the peer review findings for the internal events, internal flooding, and other PRAs that are relevant to the results of the external flooding PRA are resolved and incorporated into the development of the external flooding PRA plant response analysis. See Note XF-N-69	
XFPR-C2	INCLUDE external flood-induced failures representing failure modes of interest in the external flood PRA plant response model.	
XFPR-C3	MODEL correlation of external flooding-induced SSC failures, if applicable. JUSTIFY the correlation approach used (e.g., by demonstrating that SSC failures are correlated due to common component locations, similar hydrostatic or hydrodynamic loadings, or correlated SSC failures due to improper installation of a temporary flood barrier). See Note XF-N-70	
XFPR-C4	If screening out failure modes from further evaluation in the plant response analysis, ENSURE that at least one of the criteria of the selected approach (either deterministic or probabilistic) is met or JUSTIFY alternate screening criteria: *(a)* Deterministic approach: USE the screening criteria of SCR-3 in Table 1.10-1, subject to meeting all of the following additional criteria: Criterion A: It is not physically possible for the failure mode to impact plant systems as assessed by a demonstrably conservative, deterministic analysis; Criterion B: The failure mode does not impact a mitigation function being considered in the PRA and the failure mode does not result in (or create) another initiating event; or *(b)* Probabilistic approach: Using demonstrably conservative assessment (or a realistic assessment that meets all applicable requirements of this Section), USE the screening criteria of SCR-2 in Table 1.10-1.	

Table 4.3.13.3-4 Supporting Requirements for HLR-XFPR-C (Cont'd)

The External Flood Plant Response Analysis shall include external flood-induced SSC failures, non- external flood-induced SSC failures, unavailabilities, human errors, and multi-reactor effects that lead to release of radionuclide material. (HLR-XFPR-C)

Index No. XFPR-C	Capability Category I	Capability Category II
XFPR-C5	ADDRESS dependencies of external flood-induced failures of SSCs as part of the screening evaluation (e.g., effects of failure of a flood barrier on the failure of credited SSCs) before screening out failure modes from the plant response model.	
XFPR-C6	For new PRA logic models developed for the external flooding PRA, SATISFY the following requirements, consistent with CC-I requirements in (if applicable): *(a)* Initiating Event Analysis per the SRs of HLR-IE-A and HLR-IE-B (including initiating events associated with alternate mode operation as well as use of SSCs associated specifically with flood response, if applicable); *(b)* Event Sequence Analysis per the SRs of HLR-ES-A and HLR-ES-B; *(c)* Success Criteria Development per the SRs of HLR-SC-A (except SC-A7 which is addressed by XFPR-C11) and HLR-SC-B; *(d)* Systems Analysis per the SRs of HLR-SY-A and HLR-SY-B; *(e)* Data Analysis per the SRs of HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D. ENSURE the following are represented: *(a)* external flood-induced SSC failures; *(b)* SSC unavailabilities and failures not induced by the external flood event; and *(c)* human actions associated with plant flood response (including flood-related actions not included within the internal events model) that can give rise to risk-significant event sequences or risk-significant event progression sequences. See Note XF-N-71	For new PRA logic models developed for the external flooding PRA, SATISFY the following requirements, consistent with CC-II requirements in (if applicable): *(a)* Initiating Event Analysis per the SRs of HLR-IE-A and HLR-IE-B (including initiating events associated with alternate mode operation as well as use of SSCs associated specifically with flood response, if applicable); *(b)* Event Sequence Analysis per the SRs of HLR-ES-A and HLR-ES-B; *(c)* Success Criteria Development per the SRs of HLR-SC-A (except SC-A7 which is addressed by XFPR-C11) and HLR-SC-B; *(d)* Systems Analysis per the SRs of HLR-SY-A and HLR-SY-B; *(e)* Data Analysis per the SRs of HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D. ENSURE the following are represented: *(a)* external flood-induced SSC failures; *(b)* SSC unavailabilities and failures not induced by the external flood event; and *(c)* human actions associated with plant flood response (including flood-related actions not included within the internal events model) that can give rise to risk-significant event sequences or risk-significant event progression sequences. See Note XF-N-71
XFPR-C7	ADDRESS coexistent hazards (e.g., seismic, high winds) in the External Flood Plant Response Analysis. See Note XF-N-72	
XFPR-C8	ADDRESS conditions identified during the external flood investigation(s) (refer to the SRs of HLR-XFHA-F and HLR-XFFR-B) that are relevant to the External Flooding PRA.	
XFPR-C9	In the plant response model, for each basic event that represents an external flood-induced failure, INCLUDE the complementary "success" state where applicable to a particular SSC in cases where the external flood-caused failure probability is high and when its exclusion would affect risk insights.	
XFPR-C10	For sites with multiple reactors, INCLUDE the effects of external flooding on other reactors as it affects the reactor under study (e.g., effects on resources and organizational response, shared SSCs, and site accessibility). See Note XF-N-63, XF-N-73	

Table 4.3.13.3-4 Supporting Requirements for HLR-XFPR-C (Cont'd)

The External Flood Plant Response Analysis shall include external flood-induced SSC failures, non- external flood-induced SSC failures, unavailabilities, human errors, and multi-reactor effects that lead to release of radionuclide material. (HLR-XFPR-C)

Index No. XFPR-C	Capability Category I	Capability Category II
XFPR-C11	DEFINE mission time(s) and SATISFY the CC-I requirements in SC-A7, recognizing that mission time(s) for individual or groups of SSCs may be affected by the different phases of plant response and extended duration during which flood conditions may adversely affect the site.	DEFINE mission time(s) and SATISFY the CC-II requirements in SC-A7, recognizing that mission time(s) for individual or groups of SSCs may be affected by the different phases of plant response and extended duration during which flood conditions may adversely affect the site.
XFPR-C12	JUSTIFY the mission time(s) used by considering factors such as *(a)* the flood event duration; *(b)* the impacts of the external flood event on the site and surrounding areas (including accessibility within buildings, around the site, and in the region around the site); *(c)* the portion of the flood event duration that individual SSCs are required to operate; *(d)* the point at which the plant has reached safe and stable conditions.	

Table 4.3.13.3-5 Supporting Requirements for HLR-XFPR-D

The list of SSCs selected for External Flood Fragility Analysis shall include the SSCs that contribute to event sequences included in the External Flood Plant Response Analysis. (HLR-XFPR-D)

Index No. XFPR-D	Capability Category I	Capability Category II
XFPR-D1	USE the internal events systems model developed against Requirement SY-A1, and the internal flooding systems model, as applicable, as the basis for developing a XFEL to support the fragility analysis. INCLUDE any additional systems that may have been incorporated into the external flood event sequence model in response to XFPR-B1. See Note XF-N-67, XF-N-74	
XFPR-D2	INCLUDE in the XFEL additional SSCs that may not be explicitly modeled in the internal events and internal flood PRA model (or that may have been screened out in the internal events or internal flood PRA model), as applicable, (see Requirement XFPR-D1) that require evaluation in the external flooding PRA. See Note XF-N-74, XF-N-75	
XFPR-D3	AUGMENT the XFEL based on the review of industry flood PRA XFEL, if available or applicable to the reactor design. See Note XF-N-76	
XFPR-D4	AUGMENT the XFEL with SSCs that require evaluation in the external flooding PRA due to the effect of coexistent hazards. See Note XF-N-77	
XFPR-D5	For the SSCs identified in Requirements XFPR-D1, XFPR-D2, XFPR-D3, and XFPR-D4, IDENTIFY the failure modes of interest for the fragility analysis. See Note XF-N-78	

Table 4.3.13.3-6 Supporting Requirements for HLR-XFPR-E

The External Flood Plant Response Analysis shall include actions associated with plant response strategies (e.g., human actions to prepare for the flood event) and external flood-specific challenges to human performance. (HLR-XFPR-E)

Index No. XFPR-E	Capability Category I	Capability Category II
XFPR-E1	IDENTIFY the human failure events (HFEs) from the internal events PRA as well as those not included in existing PRA models (including preparatory and recovery actions) that are relevant in the context of the external flooding PRA. See Note XF-N-79	
XFPR-E2	For human response actions relevant to the External Flood Plant Response Analysis, SATISFY the CC-I SRs of HLR-HR-E, except where the requirements are not applicable.	For human response actions relevant to the External Flood Plant Response Analysis, SATISFY the CC-II SRs of HLR-HR-E, except where the requirements are not applicable.
XFPR-E3	For definition and specification of HFEs for human response actions, SATISFY the CC-I SRs of HLR-HR-F, except where the requirements are not applicable.	For definition and specification of HFEs for human response actions, SATISFY the CC-II SRs of HLR-HR-F, except where the requirements are not applicable.
XFPR-E4	For treatment of recovery actions, SATISFY the SRs of HLR-HR-H, except where the requirements are not applicable.	
XFPR-E5	INCLUDE HFEs in the External Flood Plant Response Analysis, such that the HFEs represent the impact of human failures at the function, system, train, or component level, as appropriate.	
XFPR-E6	For developing human error probabilities (HEPs), SATISFY the CC-I SRs of HLR-HR-G, except where they are not applicable, taking into account relevant external flood-related effects on the control room and ex-control room human actions. When addressing influencing factors and the timing considerations covered in Requirements HR-G4, HR-G6, and HR-G8, INCLUDE the effect of the external flooding hazard on the control room and ex-control room human actions. See Note XF-N-80	For developing HEPs, SATISFY the CC-II requirements of the SRs of HLR-HR-G, except where they are not applicable, taking into account relevant external flood-related effects on human actions. When addressing influencing factors and the timing considerations covered in Requirements HR-G4, HR-G6, and HR-G8, INCLUDE the effect of flood impacts on the control room and ex-control room human actions. See Note XF-N-80
XFPR-E7	EVALUATE whether credited system recoveries modeled in the internal events PRA become challenging or impossible as a result of an external flood event.	
XFPR-E8	ADJUST the credited recovery models based on results of Requirement XFPR-E7. SPECIFY the basis for recovery values, if used (e.g., based on review of procedures and assessment of conditions under which actions will be performed).	

Table 4.3.13.3-7 Supporting Requirements for HLR-XFPR-F

The analysis to quantify event sequence family frequencies shall integrate the external flood hazard, the external flood fragilities, and the external flood plant response, including uncertainties. (HLR-XFPR-F)

<table>
<tr><th>Index No. XFPR-F</th><th>Capability Category I</th><th>Capability Category II</th></tr>
<tr><td>XFPR-F1</td><td colspan="2">In the quantification of event sequence family frequencies on a plant-year basis, INTEGRATE the hazard, fragility, and systems analyses in the PRA model.</td></tr>
<tr><td>XFPR-F2</td><td colspan="2">ADDRESS overestimation of risk due to rare event approximations (e.g., where fragilities approach 1.0).
See Note XF-N-81</td></tr>
<tr><td>XFPR-F3</td><td colspan="2">ENSURE that the discretization of the hazard curves (such as the size and number of bins used to discretize the hazard curve or other numerical methods used to incorporate the hazard curve in the integration) is appropriate to demonstrate convergence of event sequence family frequencies.</td></tr>
<tr><td>XFPR-F4</td><td colspan="2">When quantifying external flood event sequence family frequencies, SATISFY the following requirements, except where the requirements are not applicable:
(a) ESQ-A4, ESQ-A6, ESQ-A7;
(b) ESQ-B1, ESQ-B2, ESQ-B3, ESQ-B5, ESQ-B6, ESQ-B7, ESQ-B8, ESQ-B9, ESQ-B10;
(c) ESQ-C1, ESQ-C2, ESQ-C3, ESQ-C4, ESQ-C5, ESQ-C6, ESQ-C7, ESQ-C8, ESQ-C9, ESQ-C10, ESQ-C11, ESQ-C12, ESQ-C13, ESQ-C14, ESQ-C15, ESQ-C16, ESQ-C17;
(d) ESQ-D1, ESQ-D2, ESQ-D3, ESQ-D5, ESQ-D6, and ESQ-D7.</td></tr>
<tr><td>XFPR-F5</td><td>USE the representative hazard function (see Requirement XFHA-E4), representative fragilities (see Requirement XFFR-D1), and a point estimate quantification of plant response model to generate point estimates of event sequence family frequencies.</td><td>QUANTIFY the mean and the uncertainties of the event sequence family frequency estimates by propagating the uncertainties associated with external flood hazard frequency, external flood fragility, and external flood plant response model events through the quantification process.</td></tr>
<tr><td>XFPR-F6</td><td colspan="2">IDENTIFY assumptions and sources of uncertainty in the External Flood Plant Response Analysis in a manner that supports Requirement XFPR-F7.</td></tr>
<tr><td>XFPR-F7</td><td colspan="2">CHARACTERIZE notable sources of uncertainty in the External Flood Plant Response Analysis (e.g., using uncertainty analysis or sensitivity studies) and SATISFY ESQ-E1 for each external flood technical subelement as identified in Requirements XFHA-E1 for External Flood Hazard Analysis, XFFR-D3 for External Flood Fragility Analysis, and XFPR-F6 for External Flood Plant Response Analysis.</td></tr>
</table>

Table 4.3.13.3-8 Supporting Requirements for HLR-XFPR-G

The External Flood Plant Response Analysis shall incorporate the data and findings of an investigation(s) to establish or confirm as-built, as-operated or as-designed, as-intended-to-operate conditions. (HLR-XFPR-G)

<table>
<tr><th>Index No. XFPR-G</th><th>Capability Category I</th><th>Capability Category II</th></tr>
<tr><td>XFPR-G1</td><td colspan="2">COLLECT information via investigation(s) about either the as-built, as-operated or as-designed, as-intended-to-operate, as applicable, plant and site characteristics relevant to the plant response analysis including consideration of operations; flooding design/licensing basis; design, construction, and performance of flood protection features; hydrology and hydraulics; and manual actions.
See Note XF-N-15, XF-N-82</td></tr>
<tr><td>XFPR-G2</td><td colspan="2">IDENTIFY potential physical interactions between SSCs, including impacts that the external flood-induced failure of an SSC may have on the function of other SSCs.
See Note XF-N-83</td></tr>
</table>

Table 4.3.13.3-8 Supporting Requirements for HLR-XFPR-G (Cont'd)

The External Flood Plant Response Analysis shall incorporate the data and findings of an investigation(s) to establish or confirm as-built, as-operated or as-designed, as-intended-to-operate conditions. (HLR-XFPR-G)

Index No. XFPR-G	Capability Category I	Capability Category II
XFPR-G3	EVALUATE the external floods plant response analysis for the as-built, as-operated or as-designed, as-intended-to-operate plant conditions via investigation(s) to do the following: *(a)* establish plant response timelines and expected plant conditions for the duration of the flood event; *(b)* ensure the accuracy of information collected from plant information sources; and *(c)* collect or confirm inputs related to: *(d)* flood scenarios, including external flood sources, external flood impact; assessments, and inleakage pathways; *(e)* SSCs located within areas that may be flooded by plant design, due to inleakage, or due to flood protection failures; *(f)* human reliability analyses; *(g)* screening. See Note XF-N-15	

Table 4.3.13.3-9 Supporting Requirements for HLR-XFPR-H

The documentation of the External Flood Plant Response Analysis shall provide traceability of the work. (HLR-XFPR-H)

Index No. XFPR-H	Capability Category I	Capability Category II
XFPR-H1	DOCUMENT the process used in the External Flood Plant Response Analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* external flood scenarios considered in the PRA; *(b)* the specific adaptations made in the internal events PRA model to produce the external flooding PRA model and the basis for those adaptations; *(c)* XFEL; *(d)* list of referenced procedures credited in the PRA; *(e)* those external flood-related influences that affect methods, processes, or assumptions used as well as the identification and quantification of the HFEs/HEPs in accordance with the SRs of HLR-XFPR-E; *(f)* the results of the External Flood Plant Response Analysis, including the major outputs of an external flooding PRA, such as event sequence family frequencies, uncertainty distributions on event sequence family frequencies (if quantified), results of sensitivity studies, and risk-significant contributors; *(g)* risk-significant contributors such as initiating events, event sequence families, SSC failures, common cause failures, and operator actions to event sequence family frequencies; *(h)* risk-significant event sequence families; *(i)* the SRs of HLR-HR-I for Human Reliability Analysis except where the requirements are not applicable; *(j)* the sources of parameter uncertainty associated with the External Flood Plant Response Analysis. See Note XF-N-84	
XFPR-H2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement XFPR-F6) associated with the External Flood Plant Response Analysis.	
XFPR-H3	For PRAs conducted on a bounding site or during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details associated with the External Flood Plant Response Analysis (as identified in Requirement XFPR-F6). See Note XF-N-21	

4.3.13.4 Peer Review Requirements for External Flooding PRA

4.3.13.4.1 Purpose

This Section states requirements for peer review of the External Flood Analysis element of the PRA. However, the initiating flood hazard may result in actions to re-configure or change the operating mode of the plant or design prior to the onset of the external flood-induced initiating event or in response to warnings. The potential for changes in plant or design configuration or mode prior to the onset of flood conditions presents unique challenges for the PRA that may require special consideration by peer reviewers.

4.3.13.4.2 Peer Review Team Composition and Personnel Qualifications

In addition to the general requirements of Section 6, the peer review team shall have collective knowledge and experience in the areas of external flood hazard assessment, flood fragility assessment, external flood investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable], and external flood plant response assessment.

4.3.13.4.3 Review of External Flooding PRA Elements to Confirm the Methodology

4.3.13.4.3.1 External Flood Hazard Analysis

The peer review team shall evaluate whether the external flood hazard study used in the PRA is appropriately specific to the bounding site covering a range of sites and has met the relevant requirements of this Standard. The peer review team shall focus on limitations associated with available data, models, and methods.

4.3.13.4.3.2 External Flood Fragility Analysis Evaluation

4.3.13.4.3.2.1 Investigation

The peer review team shall review the investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable] of the plant and bounding site, covering a range of sites to ensure the reasonableness of the findings of the analysis in terms of screening, flood impacts, SSC interactions, and the identification of critical failure parameters (e.g., height or hydrostatic and hydrodynamic loads) and modes (e.g., overtopping, exceedance of structural capacity, or inleakage). The peer review team (or peer review team representatives) shall also perform an independent walkdown, focusing on areas judged by the peer reviewers to require special attention.

For PRAs on plants prior to operation, if plant walkdown(s) is not possible, the peer review team should review the findings of the following information via interviews and reviews (e.g., tabletop reviews, computerized simulations) with engineering personnel to ensure the reasonableness of the findings of the analysis in terms of screening, flood impacts, SSC interactions, and the identification of critical failure parameters (e.g., height or hydrostatic and hydrodynamic loads) and modes (e.g., overtopping, exceedance of structural capacity, or inleakage).

4.3.13.4.3.2.2 Fragility Evaluation

The peer review team shall evaluate the analysis of SSC response to external flood hazards and whether the analysis, including data, models, and methods, meets the relevant requirements of this Standard. Specifically, the review shall focus on the hazard characterization, critical failure parameters (e.g., as identified via structural modeling), risk-significant failure mechanisms, and impacts on SSC functions (i.e., associated failure modes).

4.3.13.4.3.2.3 External Flood Plant Response Analysis

4.3.13.4.3.2.3.1 External Flood-Induced Initiating Events

The peer review team shall evaluate whether initiating events included in the external flooding PRA are properly identified and analyzed.

4.3.13.4.3.3.3.2 External Flood Event Sequence Analysis and Plant Response Models

The peer review team shall evaluate whether the plant response analysis represents the plant response to external flood, including proceduralized actions that change the plant configuration or operating mode. The peer review team shall evaluate whether the event sequences are properly analyzed and quantified in the analysis. The review team shall ensure that the XFEL is reasonable for the plant or design. When applicable, the peer review team shall focus on the construction and justification of models developed specifically for the external flooding PRA (i.e., not based on the internal events or other existing models).

4.3.13.4.3.3.3.3 Quantification Method

The peer review team shall evaluate whether the quantification method used in the external flooding PRA is appropriate for the evaluation of external flood event sequence and event sequence families. If the analysis contains screening assumptions, or assumptions that the analysis team concludes are demonstrably conservative, the peer review team shall review the reasonableness of those assumptions. The peer review team shall concentrate on the event sequence and event sequence family estimates, uncertainty bounds, and on the risk-significant contributors.

Some SRs allow for the use of assumptions and generic data. The sensitivity of the PRA results to the uncertainty associated with the use of assumptions and generic data is expected to be a focus area for the peer review.

4.3.13.5 References for External Flooding PRA

The following is a list of publications referenced in this Standard.

[XF-1] Code of Federal Regulations, Title 10, “Energy,” Part 50, “Domestic Licensing of Production and Utilization Facilities,” Appendix A, “General Design Criteria for Nuclear Power Plants,” Criterion 2, “Design Bases for Protection Against Natural Phenomena,” U.S. Nuclear Regulatory Commission, 1971

[XF-2] Code of Federal Regulations, Title 10, "Energy," Part 100, "Reactor Site Criteria," Sec. 100.10(c), "Physical Characteristics of the Site, Including Seismology, Meteorology, Geology, and Hydrology," U.S. Nuclear Regulatory Commission, 1973

[XF-3] Code of Federal Regulations, Title 10, "Energy," Part 100, "Reactor Site Criteria," Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Sec. IV(c), "Required Investigations for Seismically Induced Floods and Water Waves," U.S. Nuclear Regulatory Commission, 1973

[XF-4] Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants," Rev. 2, U.S. Nuclear Regulatory Commission, 1976

[XF-5] Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," U.S. Nuclear Regulatory Commission, 1976 (errata in 1980)

[XF-6] Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," Rev. 1, U.S. Nuclear Regulatory Commission, 1976

[XF-7] NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Sec. 2.4, "Hydrology," U.S. Nuclear Regulatory Commission, 1996

[XF-8] NUREG/CR-2300, J. HICKMAN et al., "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," American Nuclear Society/Institute of Electrical and Electronic Engineers/U.S. Nuclear Regulatory Commission, 1983

[XF-9] NUREG/CR-5042, UCID-21223, C. Y. KIMURA and R. J. BUDNITZ, "Evaluation of External Hazards to Nuclear Power Plants in the United States," Lawrence Livermore National Laboratory/U.S. Nuclear Regulatory Commission, 1987

[XF-10] NUREG/CR-2815, BNL-NUREG-51559, Vol. 2, M. McCANN, J. REED, C. RUGER, K. SHIU, T. TEICHMANN, A. UNIONE, and R. YOUNGBLOOD, "Probabilistic Safety Analysis Procedures Guide," Brookhaven National Laboratory/U.S. Nuclear Regulatory Commission, 1985

[XF-11] NUREG/CR-5477, R. J. BUDNITZ and H. E. LAMBERT, "An Evaluation of the Reliability and Usefulness of External-Initiator PRA Methodologies," Future Resources Associates, Inc./U.S. Nuclear Regulatory Commission, 1990

[XF-12] H. LIU, Wind Engineering: A Handbook for Structural Engineers, Prentice Hall, 1991

[XF-13] Estimating Probabilities of Extreme Floods: Methods and Recommended Research, National Research Council (U.S.) Committee on Techniques for Estimating Probabilities of Extreme Floods, National Academy Press, Washington D.C., 1988

[XF-14] Feasibility of Assigning a Probability to the Probable Maximum Flood, prepared by Work Group on Probable Maximum Flood Risk Assessment, under the direction of the Hydrology Subcommittee of the Interagency Advisory Committee on Water Data, U.S. Office of Water Data Coordination, 1986

[XF-15] NUREG/CR-4710, SAND86-1797, G. A. SANDERS, D. M. ERICSON, Jr., and W. R. CRAMOND, "Shutdown Decay Heat Removal Analysis of a Combustion Engineering 2-Loop Pressurized Water Reactor: Case Study," Sandia National Laboratories/U.S. Nuclear Regulatory Commission, 1987

[XF-16] E. H. VANMARKE and H. BOHNENBLUST, "Risk-Based Decision Analysis for Dam Safety," R82-11, Massachusetts Institute of Technology, Department of Civil Engineering, 1982

[XF-17] M. W. McCANN, Jr., and G. A. HATEM, "Progress on the Development of a Library and Data Base on Dam Incidents in the U.S.," Stanford University Department of Civil Engineering, Progress Report No. 2 to Federal Emergency Management Agency; available in an alternative form as G. A. HATEM, "Development of a Database on Dam Failures in the United States: Preliminary Results," Engineering Thesis, Stanford University Department of Civil Engineering, 1985

[XF-18] M. W. McCANN, Jr., and A. C. BOISSONNADE, "Probabilistic Flood Hazard Assessment for the N Reactor, Hanford, Washington," UCRL-2106, Lawrence Livermore National Laboratory, 1988

[XF-19] NEI 12-07 (Rev. 0), "Guidelines for Performing Verification Walkdowns of Plant Flood Protection Features," Nuclear Energy Institute, May 2012

[XF-20] ASME/ANS RA-Sb-2013 (R2019), "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," American Society of Mechanical Engineers and American Nuclear Society, 2013

[XF-21] NRC, Information Notice 2015-01: "Degraded Ability to Mitigate Flooding Events," 2015

NONMANDATORY APPENDIX XF: NOTES AND EXPLANATORY MATERIAL FOR EXTERNAL FLOODING PRA

XF.1 NOTES ASSOCIATED WITH EXTERNAL FLOODING PRA

Table XF-1 Notes Supporting External Flooding PRA Requirements

Number	Notes
XF-N-1	The description of a bounding site can be the identification of an existing site if that site bounds the hazard for other sites under consideration. See XFHA-A1
XF-N-2	This requirement is intended to ensure that all relevant hazard mechanisms and combinations of mechanisms are identified. A broad range of natural (e.g., weather-induced) and man-made (e.g., caused by operational releases from dams) phenomena may lead to flooding at nuclear power plant sites. Flooding events are generally categorized by flood-causing mechanisms, which include the following: *(a)* LIP as well as the associated site and roof drainage; *(b)* flooding from streams and rivers; *(c)* dam releases and failures (including operational releases as well as hydrologically induced, seismically induced and other failure mechanisms) as well as failure of on-site storage basins and impoundments (if not addressed by a different PRA; e.g., internal flood PRA); *(d)* surge; *(e)* seiche; *(f)* tsunami; *(g)* ice effects; *(h)* channel migrations and diversions. In addition to site flooding arising from the occurrence of the individual flood-causing mechanisms described above, combinations of flood-causing mechanisms may also lead to site flooding. Examples of potentially relevant combinations of flood mechanisms may include (but are not limited to) the following: *(a)* river floods with concurrent site precipitation and wind-generated waves; *(b)* basin-wide precipitation along with snowmelt, leading to river flooding; *(c)* seismic dam failures with concurrent river flood; *(d)* river flooding concurrent with storm surge event; *(e)* storm surge events concurrent with high winds and precipitation; *(f)* high water level concurrent with seiche. A systematic approach (e.g., logic structures) will be useful in identifying relevant combinations of mechanisms (e.g., low severity river flooding combined with high severity wind-generated wave effects or vice versa as well as combinations involving moderate severity river flooding and winds). The intent of this requirement is to identify all applicable flood mechanisms and combinations, but not to screen out mechanisms or combinations. Screening is addressed in subsequent SRs. Treatment of combinations of flood mechanisms (in the absence of coexistent hazards; XFHA-A3) does not typically require interactions with other technical elements of this Standard. Treatment of coexistent hazards will involve interactions with other technical elements of this Standard (see commentary associated with XFHA-A3). See XFHA-A2

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-3	Coexistent hazards include secondary and concurrent hazards (defined in the Section 2.2). An example of a secondary hazard is an external flood from a dam failure caused by a seismic event (or, similarly, a tsunami following a seismic event) when the seismic event may also adversely affect the site, including causing damage to site flood protection. An example of a concurrent hazard is high winds occurring concurrent with a storm surge event, both of which are caused by a hurricane. Another concurrent hazard may be a moderate wind event occurring concurrent with a large rainfall event caused by a thunderstorm. Treatment of coexistent hazards generally requires interactions with other Sections of the PRA Standard. Coexistent hazards could exacerbate the impact of the flood on the plant. This requirement is intended to ensure the analyst has thought about the impact of coexistent hazards before screening a flood hazard mechanism. For example, if the existence of passive flood protection is credited as part of a screening of a flood events likely to involve high winds, this SR requires that the impact of the coexistent hazard (e.g., impacts of high winds concurrent with the flood event) be considered in the screening analysis. See XFHA-A3
XF-N-4	This nonmandatory appendix note number is intentionally blank.
XF-N-5	Examples of relevant information may include (but is not limited to) historical, regional, and site-specific weather and flood history; flood characteristics and changes to the plant watershed; upstream flood features such as dams (e.g., dam condition, design, operating instructions, and reservoir and upgrade history); and site and regional topography and bathymetry. See XFHA-A4, XFHA-D1
XF-N-6	Requirement XFHA-C8 addresses situations in which information cannot be obtained for particular dams. See XFHA-A4
XF-N-7	This SR refers to use of up-to-date information, which means that data collected is expected to be reasonably current at the time the probabilistic flood hazard analysis is performed. Existing information may be used with the caveat that the information has been judged to be generally representative of the present state-of-knowledge. For additional clarification regarding use of up-to-date information, see commentary associated with HLR-XFHA-D. XFHA-C6, XFHA-C7, XFHA-C8, XFHA-C9, and XFHA-C10 provide mechanism-specific considerations that may require information collection to support treatment of each flood hazard mechanism. See XFHA-A4
XF-N-8	This nonmandatory appendix note number is intentionally blank.
XF-N-9	Screening is intended to remove from further consideration "obvious scenarios" that are not consequential to the plant. In general, if detailed analysis is required to support screening, then screening is likely not appropriate. As such, this SR refers to a conservative assessment or a realistic assessment that meets all requirements of External Flooding PRA. This is intended to represent that screening assessments should be "obvious and conservative"— if the analysis is not "obvious and conservative," then the SRs of External Flooding PRA should be applied as part of an External Flooding PRA (even if simplified in overall scope). The LIP flood hazard mechanism is unique in that "rains falls on every site." Thus, to screen the LIP mechanism from the PRA it is important that the screening analysis demonstrate that the LIP flood hazard mechanism is not a risk-significant contributor. Deterministic screening of the LIP hazard is focused on confirming that *(a)* a demonstrably bounding LIP hazard be constructed (e.g., worst credible antecedent conditions, atmospheric conditions, and worst credible rainfall rates and storm distributions); *(b)* the demonstrably bounding deterministic LIP hazard (typically characterized by site ponding depth) does not exceed (or otherwise challenge) the plant passive protection and that no cliff-edges exist immediately beyond the capacity of flood protection.

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-9 (Cont'd)	The deterministic screening approach is a blended qualitative and quantitative approach but does not require quantification of hazard frequency. The focus of the assessment is to confirm there is no challenge to the site passive protective barriers under the conservatively defined hazard and that no cliff-edge effects exist immediately beyond the capacity of flood protection (e.g., flood mechanisms with a small margin between a conservatively defined hazard and the capacity of flood protection would not meet this criterion). When considering the potential for a flood to adversely affect flood risk-significant SSCs, it is expected the flood height (or elevation), associated effects, and flood event duration be considered. The probabilistic focuses on use of bounding/conservative hazard frequencies for screening, but also considers the reliability of site protection and mitigation capabilities) and overall risk insights from flooding hazards. Even when conservative assessments are performed, it is important that epistemic uncertainties be addressed (e.g., it is considered insufficient to use a single statistically based method for estimating the frequency of consequential flooding for the purposes of screening a flood hazard). If focusing on hazard frequency for meeting Criterion A to screen the LIP flood hazard mechanism, it is expected that the analysis will use the conservatively estimated frequency of flood hazard events associated with the mechanism, resulting in adverse plant effects (e.g., flood above the lowest power block elevation). It is noted that assigning frequencies solely to stylized deterministic event combinations does not result in hazard frequency estimates that are meaningful or appropriate for use in screening assessments. The conservative deterministic analysis may include different elevations or depths at different locations around the site. When ensuring that the impacts of the LIP hazard do not exceed the capacity of passive flood protection (deterministic Criterion B), the capacity of the flood protection should be compared against the LIP impacts at the location of the flood protection. If focusing on hazard frequency for meeting Criterion A to screen the flood hazard mechanism, USE the conservatively estimated frequency of flood hazard associated with the mechanism resulting in adverse plant effects (e.g., flood above the lowest power block elevation). See XFHA-A5
XF-N-10	The commentary associated with XFHA-A5 provides a discussion of screening approaches in the context of LIP hazards. In general, similar considerations apply with this SR, which focuses on other flood hazard mechanisms. However, it is noted that the criteria associated with hazards other than LIP are generally more restrictive, which is intended to represent the increased damage potential associated with these mechanisms. For example, when screening hazards other than LIP, Criterion A is intended to preclude screening of any mechanisms that create a potential for water to impinge on SSCs contained in the power block or that are otherwise considered flood response SSCs. However, it is noted that geographic considerations may be appropriate for use in screening certain mechanisms. For example, it may be appropriate to exclude from further consideration the impacts of hurricane-induced surge or tsunami effects far from a body of water. The SR is intended to allow for these types of justifications. If focusing on hazard frequency for meeting the probabilistic approach criterion to screen the flood hazard mechanism, the hazard frequency should be applied to the "impact threshold" [e.g., the lowest of (a) the power block elevation or (b) the elevation of flood response SSCs that could be exposed to floodwaters or water ingress] using a conservatively estimated frequency of flooding. Flood protection features (whether active or passive) should not be credited as part of the deterministic screening. See XFHA-A6

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-11	The historical database for tsunamis extends for several hundred years in both the Pacific and Atlantic Ocean basins, with less reliable historical data going back somewhat further. Given a distant tsunami arriving at a specific location, it is feasible to determine how large the tsunami-induced flood will be, considering the local offshore subsurface topography. Usually, an engineering analysis is sufficient to screen out tsunamis. If a site-specific probabilistic (numerical) analysis of the hazard frequency is required, the uncertainties are often large and therefore must be accounted for properly. See XFHA-A6
XF-N-12	In the ASME/ANS RA-Sb-2013 (R2019) [XF-20] requirement from which this requirement was extracted, the user was directed to ensure that the total risk of all screened floods was not greater than the value listed of SCR-1. While this approach is feasible for plants using risk surrogates such as core damage frequency, this requirement becomes much more difficult and burdensome to achieve for an integrated Level 3 PRA as used in this Standard. As a result, the requirement was modified to ensure that event sequence family frequencies were not screened by SCR-1, which is consistent with the overall screening philosophy used in this Standard. See XFHA-A7
XF-N-13	See also Requirement XFHA-A4. Several generic databases exist on U.S. dam failures, categorized by the different dam types (earthfill dams, concrete dams, etc.). See [XF-16] and [XF-17]. These databases must be used with care, depending on how closely the specific dam fits into the database. The mean failure rate for all U.S. dams is in the range between ~10^{-4}/yr and ~10^{-5}/yr [XF-9]. However, for some modern dams with extensive engineering, values <10^{-5}/yr have been quoted [XF-18], while for older, poorly constructed dams, values near 10^{-3}/yr could be appropriate. An accurate and useful probabilistic analysis of any specific dam would require detailed engineering evaluations. XFHA-A7 is intended to represent the importance of ensuring that the aggregate risk from all flood mechanisms is not significant when considering the baseline risks of the plant (e.g., ensure there are not a large number of flood mechanisms that "barely screen" individually such that their aggregate contribution may be important). See XFHA-A7
XF-N-14	XFHA-A8 requires that investigations and a review of plant records have been performed and that the review did not indicate any relevant deficiencies with respect to as-built, as-operated or as-designed as-intended-to-operate configuration of the plant or that relevant deficiencies have been otherwise addressed as part of the screening process (e.g., the conclusion of the screening is not affected by the deficiency). Consideration as part of the screening process may result in the conclusion that the deficiency will not affect the results of the PRA (e.g., justification that the deficiency will be resolved/corrected prior to completion of the PRA). "Relevant deficiencies" refer to deficiencies that are relevant in the context of the screening conclusions and may include degraded condition of flood diversion features, flood protection, barriers, or drainage systems. The plant information noted in this SR may include documents such as corrective action program entries. See XFHA-A8
XF-N-15	Examples of investigations include, but are not limited to, actives such as tabletop review(s), physical/computerized walkdown(s) of the plant, and/or interview(s) with knowledgeable personnel. These investigation(s) may need to be supported by a review of plant documents, especially in the context of external hazard evaluations. This SR does not imply that each of these activities need to be performed to meet the SR. Further, if appropriate, one investigation can meet the intent of multiple investigation SRs if the investigation is scoped and structured appropriately. See XFHA-A8, XFHA-F1, XFFR-B1, XFFR-B2, XFPR-G1, XFPR-G3

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-16	XFHA-A9 requires that hazard mechanisms identified in XFHA-A3 that are not screened out by the above processes be identified. The SR specifically includes consideration of credible combinations of events. Examples of combinations of mechanisms include the following: *(a)* "relevant deficiencies" refers to deficiencies that are relevant in the context of the screening conclusions and may include degraded condition of flood diversion features, flood protection, barriers, or drainage systems; *(b)* the plant information noted in this SR may include documents such as corrective action program entries. See XFHA-A9
XF-N-17	XFHA-B1 recognizes that the flood hazard function will typically represent flood hazard severity using one or more measures such as flood height or elevation. However, this SR also recognizes that other factors are important and relevant to the characterization of a flooding hazard. Other factors may include static and dynamic loading and consideration of debris transport and impact. This SR is intended to require that these aforementioned factors be addressed but offer flexibility in how that is done. The goal of this SR is to ensure flood scenarios are properly characterized for probabilistic treatment. Characterization typically includes factors causing significant plant challenges such as overflow of barriers, static and periodic water surface elevations (waves), water-induced stress loading on barriers (static and dynamic loads), and debris impacts. In addition, flood event duration is an important characterization of a flooding hazard and may differ based on flood severity (e.g., less severe, but consequential, flood events may be associated with less warning time than more severe flood events). See XFHA-B1
XF-N-18	For reactors in the pre-operational stage, the bounding site from XFHA-A1 should be used for all SRs in HLR-XFHA-B. See XFHA-B1
XF-N-19	XFHA-B2 recognizes that the severity and characteristics of flooding may differ across the site. For example, during a storm surge event, wave effects may occur primarily on seaward sides of structures. As another example, during LIP events, the depth and duration of water ponding and the relationship of the ponding to incipient building flooding may vary locally depending on the site topography. See XFHA-B2
XF-N-20	For PRAs performed during the pre-operational stage, include proposed changes to the topography and assumed placement of buildings. See XFHA-B2
XF-N-21	This SR is not applicable to operating plants. See XFHA-B2, XFHA-B3, XFHA-G3, XFFR-F3, XFPR-H3
XF-N-22	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See XFHA-B3
XF-N-23	The basis for the grouping should consider flood height, associated effects, and flood event duration (e.g., hazards should only be grouped if there are sufficient similarities in all measures of flood severity). See XFHA-B4

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-24	A range of methods exist to predict annual exceedance frequencies for floods associated with various hazard mechanisms. Methods such as those focusing on extreme value analysis of observational data (often referred to as "flood frequency analysis") may be less reliable at lower frequencies and may not be supported by the existing state-of-practice. Probabilistic approaches that leverage observational data as well as numerical, physical, or other process models may be more readily applied to estimate hazards with lower frequencies. Such potentially applicable approaches include joint probability analyses (e.g., the joint probability method for storm surge) and stochastic simulation. These SRs are not prescriptive in identifying the methods that are most applicable to a particular site. Instead, these SRs require that modeling methods and assumptions be identified and justified. Relevant considerations in developing this justification may include the expected significance of the hazard mechanism to the plant risk, the degree to which the selected method(s) accommodates treatment of uncertainties, the availability of data, and the degree to which the selected method(s) is supported by the existing state-of-practice for the annual exceedance frequencies of relevance. CC-I requires justification of demonstrably conservative overall assessment (even if not all assumptions are conservative or bounding). CC-II applications are encouraged to use realistic assumptions to facilitate computing a suite of hazard functions (e.g., curves) from which a mean and other facilities hazards can be derived. If conservative or bounding assumptions are used in Category II applications, these SRs are intended to ensure such assumptions do not significantly distort risk insights. This SR focuses primarily on the development of the model for representing aleatory variability. See XFHA-C2
XF-N-25	For XFHA-C3, CC-I requires that, when statistical methods (e.g., flood frequency approaches that fit extreme value distributions to flood parameters and extrapolate to develop hazard curves) are used in developing the probabilistic flood hazard analysis, the analyst provide justification for the method, selection, use, and range of applicability. Many statistical approaches have been developed for the purposes of estimating hazards associated with a limited range of exceedance frequencies. Limitations regarding the technical defensibility of extrapolation using statistical extreme value analysis methods may result in a need to treat hazards considering a combination of mechanistic treatment of hazards, statistical analysis, and engineering judgments. For XFHA-C3, both CC-I and CC-II require the analysis provide a basis for the selected models and methods (e.g., provide an explanation that the selected models and methods are consistent with the state-of-practice, when appropriate). Epistemic uncertainties associated with model and method selection are considered in the associated treatment of uncertainties (which is a focus of HLR-XFHA-E and associated SRs). Relevant epistemic uncertainties may include (but are not limited to) method and model selection, distribution assumptions, and data selection and filtering. See XFHA-C3
XF-N-26	Inevitably, the methodology used will go beyond the data that was used to develop the probabilistic flood hazard analysis, but these SRs are intended to ensure the extrapolation approach is justified and performed using a defined process. This is intended to ensure that the extrapolations are reasonable and consider phenomena- and site-specific issues (e.g., regulated watersheds, limited data records, ungauged locations, and historical or paleoflood data). Uncertainty in extrapolation is treated within the consideration of epistemic uncertainty, which is a focus of HLR-XFHA-E and associated SRs. In particular, consideration of phenomenological and meteorological considerations may be necessary to augment statistical analysis methods. See XFHA-C4

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-27	This SR specifies that the hazard function (e.g., hazard curve) extend to sufficiently low values of exceedance frequency such that the risk of the hazard may be appropriately established. For example, truncation of the tails of the distribution used to develop the hazard function may unintentionally exclude risk-significant scenarios. This SR also cautions the modeler to look at potential cliff-edge effects when performing the truncation. See XFHA-C5
XF-N-28	The bulleted list provides a high-level checklist of issues to be considered in developing hazard functions and/or performing probabilistic flood hazard analyses for LIP. The applicability of each issue may depend on the methods applied in performing the probabilistic flood hazard analysis. See XFHA-C6
XF-N-29	These SRs are intended to provide mechanism-specific considerations. These SRs state that the factors listed should be addressed or identify that the factor is not applicable or consideration is not necessary (e.g., because hazard results are not sensitive to the factor or consideration of the factor would not result in consequential impacts). There may be mechanisms for which a hazard analysis is needed but for which "mechanism-specific requirements" are not provided in this Standard (e.g., river diversion). The lack of mechanism-specific requirements is not intended to imply that no hazard analysis is needed. The lack of requirements for these "other mechanisms" represents the limited state-of-practice regarding those other mechanisms. Nonetheless, the other requirements (that are not mechanism-specific) still apply. See XFHA-C6, XFHA-C7, XFHA-C8, XFHA-C9, XFHA-C10, XFHA-C11
XF-N-30	The bulleted list provides a high-level checklist of issues to be considered in developing hazard functions and/or performing probabilistic flood hazard analyses for riverine floods with no upstream dams. The applicability of each issue may depend on the methods applied in performing the probabilistic flood hazard analysis. It is noted that multiple methods may be employed depending on the frequency range of interest (e.g., generalized extreme value analysis, or mechanistic treatment of hazard phenomena using numerical models and stochastic sampling). See XFHA-C7
XF-N-31	The bulleted list provides a high-level checklist of issues to be considered in developing hazard functions and/or performing probabilistic flood hazard analyses for riverine floods with upstream dams. These considerations also apply to on-site water impoundments. The applicability of each consideration may depend on the methods applied in performing the probabilistic flood hazard analysis. When considering scenarios involving failures of upstream dams, the analysis may involve mechanistic treatment of dam failures (e.g., through use of numerical models). It is noted that site flooding may be caused by operational releases from dams and need not involve failure of a dam (i.e., uncontrolled release). This SR also notes that some dams may be screened from further considerations based on a determination that its failure (alone or in combination with other dams) will not be consequential to the site. It is also noted that the list includes treatment of downstream dams as a relevant consideration because failure of downstream dams may affect flood severity by releasing water and lowering flood heights. This SR also allows for situations in which information cannot be obtained for particular dams (e.g., due to information access restrictions). See XFHA-C8

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-32	The bulleted list provides a high-level checklist of issues to be considered in developing hazard functions and/or performing probabilistic flood hazard analyses for pressure-induced phenomena (e.g., seiche, extratropical storms, hurricanes). The applicability of each consideration may depend on the methods applied in performing the probabilistic flood hazard analysis. This requirement refers to "synthetic data," which is meant to refer to data that has been generated via models embodying physics of the natural processes to create synthetic storm "histories," rather than the direct use of historical observations. It is noted that the use of numerical models to develop synthetic data is associated with assumptions and uncertainty. See XFHA-C9
XF-N-33	The bulleted list provides a high-level checklist of issues to be considered in developing hazard functions and/or performing probabilistic flood hazard analyses for tsunamis or seismic-induced seiche flood events. The applicability of each consideration may depend on the methods applied in performing the probabilistic flood hazard analysis. It is noted that estimation of hazard frequencies (and some other elements of the analysis; e.g., ground mass movement) should have been performed as part of the analysis to support a seismic PRA. See XFHA-C10
XF-N-34	Coexistent hazards may include events that are caused by the same phenomena. For example, *(a)* wind impacts in conjunction with flooding induced by storm surge or wind impacts concurrent with a precipitation event; *(b)* tsunamis caused by seismic events that may also impact plant SSCs, which are considered later in the fragility and plant response portions of the PRA. Coexistent hazards may also include combinations of phenomena that may be caused by independent phenomena. For example, a relatively frequent wind event (e.g., ambient conditions associated with 2-yr wind speed) may occur coincidently with a relatively infrequent riverine flooding event that may persist for significant period of time (e.g., an event on an upstream watershed or a large operational release from a dam leading to site flooding that persists for several months). It is noted that these relatively frequent, lower intensity wind events are not generally covered by the requirements of High Winds PRA and thus must be treated within the context of External Flooding PRA. See XFHA-C11
XF-N-35	For reactors in the pre-operational stage, the bounding site from XFHA-A1 should be used for all SRs in HLR-XFHA-D. See XFHA-D1
XF-N-36	When extrapolating frequencies to long return periods, it is important to understand the extent to which the extrapolation is consistent with available data and the state-of-practice. This requirement recognizes that extensive data sets (e.g., regional and paleoflood data) may not be applicable or available for all sites or hazard mechanisms. However, when this additional data is available and applicable, this SR specifies that it should be used as part of a CC-II assessment (use of this information is optional in a CC-I assessment). See XFHA-D1
XF-N-37	Examples of model parameters will depend on the approach used. For example, under data-driven approaches, parameters may relate to probability distribution parameters or parameters related to data filtering. For mechanistic models, parameters may include specific numerical values, decisions required to execute numerical models, characteristics of meteorology, as well as antecedent or initial conditions (e.g., parameters affecting infiltration). XFHA-C6 through XFHA-C10 provide flood hazard mechanism-specific considerations that may require information collection. See XFHA-D3

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-38	Information regarding river operations and history will typically be available from the dam operator. However, special agreements may be necessary to obtain information related to dams regulated by government agencies. See XFHA-D4
XF-N-39	Aleatory variability and epistemic uncertainties are defined in the glossary of this Standard (Section 2.2). Aleatory variability is typically represented by probability distributions (e.g., distributions on storm or snowmelt parameters) and expressed as a hazard function (e.g., hazard curve). Epistemic uncertainty is typically expressed by including various technical interpretations (e.g., alternate data sources, options for filtering data, or alternate functional forms for probability distributions) and developing multiple hazard functions or estimates from which a mean hazard function and fractiles can be derived. Relevant epistemic uncertainties may include (but are not limited to) method and model selection, probability distribution assumptions, and data selection and filtering. Treatment of epistemic uncertainties may involve iterative processes and sensitivity studies to identify important sources of uncertainties. See XFHA-E2, XFHA-E3
XF-N-40	For CC-I, the intent of this SR is to allow the analyst to develop a hazard function that provides a "representative" estimate of hazard frequencies or bounding values. "Representative" estimates should aim to be realistic but may include conservative biases. It is noted that the hazard function need not be a formal mathematical formula (e.g., in some applications, dam failure frequencies may be estimated using techniques such as event trees). In general, this SR requires that the approach selected and the interface with the plant response analysis be justified. The "hazard function" is often a "hazard curve;" however, a discontinuous function may be appropriate (e.g., when considering dam failures with simplifying assumptions regarding breach parameters). CC-II requires calculation of a mean hazard function. This includes the following: *(a)* development of hazard functions for each flood mechanism identified in XFHA-A8 (e.g., development of separate hazard functions for riverine, dam failure-induced flooding, and LIP); or *(b)* aggregation of contributions from multiple mechanisms through development of a composite flooding hazard function (e.g., development of a single hazard function that includes the contributions of riverine, dam failure-induced flooding, and LIP); or *(c)* a combination of approaches (a) and (b) (e.g., development of one hazard function that includes contributions from riverine and dam failure-induced flooding, and development of another hazard function for LIP). Additional commentary regarding each of the above items is provided below: *(a)* When using mechanism-specific hazard functions in mechanism-specific PRAs, aggregation of results will occur when calculating flooding event sequence frequencies. This approach will typically be appropriate when different mechanisms have different characteristics (e.g., associated effects or flood event durations), different effects on the site, or are associated with different site response (e.g., different flood response strategies for different mechanisms). For example, under this approach, external flood PRAs may be developed separately for dam failure and for LIP, which would necessitate development of hazard curves for each mechanism. *(b)* Approaches involving composite hazard functions that aggregate the contribution from multiple flood mechanisms into a single hazard function are expected to be used most frequently when using conservative approaches in which a single flood parameter (e.g., flood height) is sufficient to characterize the aggregated mechanisms and plant response to the flooding mechanisms is similar. For example, under this approach, a single hazard curve may be developed to represent the hazard from both riverine flooding (without dam failures) and dam failure-induced flooding.

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-40 (Cont'd)	*(c)* A combination of *(a)* and *(b)* might be appropriate when a group of mechanisms are handled via a composite hazard functions and other mechanisms are handled individually. In defining composite hazard functions [option *(b)* above], it may be necessary to ensure that the combined hazard appropriately develops the hazard information needed for the plant response and fragility analyses so that an appropriate risk profile can be developed. See XFHA-E4
XF-N-41	This SR is not applicable to PRAs performed for a specific site. See XFHA-G4
XF-N-42	Although a site-specific and plant-specific analysis of the fragilities of SSCs provides the most realistic estimates, for the pre-operational stage PRA, it may be acceptable to develop fragility estimates on some other basis (e.g., generic information), provided that the uncertainties introduced are acceptable for the application. For PRAs performed during the pre-operational stage, decoupling the fragility analysis from the hazard analysis to the extent practical allows for the fragility analysis to bounding a range of sites. It may be necessary to use generic data for this purpose. See XFFR-A1, XFFR-A3, XFFR-A4, XFFR-A5, XFFR-B5, XFFR-C1, XFFR-C2, XFFR-D1, XFFR-D2, XFFR-D3, XFFR-E2
XF-N-43	The XFEL is primarily developed as part of HLR-XFPR-D of the plant response model technical element. Many of the SSCs for which a fragility analysis will be needed will come from the internal events PRA model, since (a) that model evaluates those SSCs whose random failure contributes to event sequence family frequencies, and (b) a fraction of those SSCs will be susceptible to external flood damage. However, some SSCs whose flood-induced failure would contribute to event sequence family frequencies have not been considered in the internal events PRA, in particular when their random failure probability is very low compared to their conditional probability of failure given an external flood impact. This would be the case, for example, for walls or doors, which typically will not fail randomly with an appreciable probability but might fail to perform their function when exposed to the loads associated with an external flood event. The identification of these SSCs and the confirmation that they may be of concern requires an investigation, as further discussed under HLR-XFFR-B and its SRs. In addressing this SR, the list of SSCs obtained should be comprehensive and include SSCs identified as potentially requiring a fragility analysis, but that may subsequently be screened out under XFFR-A2 and XFFR-A4. See XFFR-A1
XF-N-44	An example of chemically or physically incompatible fluid impacts is sodium, which reacts exothermically with water. If a holding tank of activated sodium leaks during an internal flood scenario, the impact of released energy and hydrogen gas generation from this reaction should be assessed. See XFFR-A2
XF-N-45	The documentation of the screening methodology, including the basis for the screening of SSCs, is necessary for traceability and will facilitate peer review and subsequent PRA updates that might lead to a reevaluation of these SSCs. Under the probabilistic approach, this SR refers to a conservative assessment or a realistic assessment that meets all requirements of external flood. This is intended to represent that screening assessments should be "obvious and conservative"— if the analysis is not "obvious and conservative," then the SRs of external flood should be applied as part of an external flooding PRA (even if simplified in overall scope). See XFFR-A3
XF-N-46	This SR complements XFFR-A1 and ensures that SSCs not typically included in the internal events PRA but that are otherwise key to the external flooding PRA (e.g., flood walls) are included in the fragility assessment. This includes temporary SSCs, such as flood barriers (e.g., seals) and equipment (e.g., mobile pumps). See XFFR-A4

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-47	The effects of coexistent hazards require special consideration since coexistent hazards could lead to the failure of SSCs otherwise intrinsically robust when only exposed to an external flood. For example, a seismic event could cause damage to a wall and prevent it from performing its protection function during a subsequent flooding event (e.g., in case of seismically induced dam failure) that a flooding event alone would not have caused. This is further discussed under HLR-XFFR-E and its SRs. In other words, the intent of this SR is to ensure that potential adverse impacts of coexistent hazards are taken into account in the screening evaluation of SSCs. See XFFR-A5
XF-N-48	XFFR-B1 refers to the performance of an investigation to collect information about the as-built, as-operated plant or as-designed, as-intended-to-operate site characteristics relevant to the fragility analysis. In some cases, this will require a new investigation; however, a previous investigation may have been performed that can be leveraged (e.g., walkdowns performed to support post-Fukushima activities). When using existing prior investigation information for the external flooding PRA, confirmation is needed to ensure that the PRA model provides an appropriate representation of the current plant condition (e.g., represents any observed degradation of SSCs) and configuration, which may have evolved since the performance of the previous walkdown. For example, a "walk-by" of SSCs inspected during the prior investigation may be sufficient if a walk-by can confirm the ongoing adequacy of the information previously gathered. This requirement refers to current condition and configuration. The use of the word "current" is not intended to require continuous checking for and updating of information throughout the processing of performing the PRA over a reasonable time period. Instead, this requirement is intended to require that analysts use appropriately up-to-date information about the plant condition and configuration at the time analysis is started, which is often referred to as the "model freeze date" or "model cutoff date." To support the investigation, the following list of items (adapted from Section 4.2 of NEI 12-07 [XF-19]) can be taken as a starting point for consideration in the identification of flood protection features: *(a)* determine current site topography and any changes that may have affected the topography assumed by the licensing basis flood evaluation and other previously performed evaluations (e.g., post-Fukushima or periodic reevaluation); *(b)* determine changes to site building elevations and site configurations including buildings that have been added or modified or significant changes to land use (e.g., additional paved areas) since the current licensing basis flood evaluation (and other previously performed flood evaluation) was completed; *(c)* determine the barriers important to resisting the effects of external flooding (e.g., structures, walls, floors, doors, etc.); *(d)* identify penetrations through barriers, such as trenches and cable openings that could provide a path for flood water to enter buildings and the means used to seal these penetrations; *(e)* identify instrumentation relied upon to detect water in rooms and the associated warning system; *(f)* identify any features or pathways credited for flood water relief (e.g., surface drainage swales, subsurface drainage system, culverts, floor/yard drains, etc.); *(g)* review plant response procedures for external floods and identify any incorporated or exterior equipment that is credited for flood protection or mitigation; *(h)* identify any situations for which temporary plant equipment (e.g., portable pumps, sandbags, temporary barriers, etc.) is credited to protect or mitigate the effects of the external flooding event. See XFFR-B1
XF-N-49	XFFR-B2 is aimed at ensuring that the walkdown records the actual configuration of the plant, as well as the associated condition of SCCs (including any observed degradation). See XFFR-B2

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-50	This SR is likely not applicable to PRAs performed during the pre-operational stage. See XFFR-B2
XF-N-51	XFFR-B3 is intended to require that the investigation consider more than just the flood height; i.e., consider the associated effects and the flood event duration. The consideration of flood event duration represents the understanding that adverse impacts on SSC integrity can occur during the flood event duration (e.g., due to extended exposure to hydrostatic pressures, repeated wave overtopping of a barrier, or persistent inleakage through/around seals). See XFFR-B3
XF-N-52	The identification of physical or spatial interactions is a key attribute of investigation. The emphasis here is on identifying external flooding impacts from flood height, associated effects, and flood event duration (per XFFR-B3). For example, physical interactions may include the effect of the potential undermining of foundations on the structure or equipment contained within it (as relevant, given the mission times of the PRA), and the effects of failure of one component on other components. See XFFR-B4
XF-N-53	XFFR-B5 simply expresses that the information gathered during the walkdown be both documented and used in the development of fragilities (via HLRs HLR-XFFR-C and HLR-XFFR-D, and their SRs). See XFFR-B5
XF-N-54	XFFR-C1 "links" the fragility analysis and the plant response model. The action referenced in this SR involves identification of external flood impacts on SSCs in terms of failure mechanisms. Examples of failure mechanisms include inundation from leakage of doors and seals, overturning of the SSC caused by submergence, and structural failure due to impact from debris. It is noted that different flood mechanisms may lead to different failure mechanisms. In turn, failure mechanisms of SSCs are assigned to the failure modes considered in the plant response model. An example of a CC-I conservative approach for identification of failure mechanisms may include assumptions of failure as soon as water impinges upon an SSC (i.e., as soon as it "gets wet"). CC-II approaches may involve more realistic assumptions regarding failure (e.g., the ability of an SSC to withstand limited impingement of floodwaters). See XFFR-C1
XF-N-55	This SR is intended to require that the fragility analysis consider not just flood height (elevation), but also associated effects and flood event duration. See XFFR-C2
XF-N-56	The term "fragility" is used in this Standard and is typically represented with a fragility function (or curve). The fragility function could, in principle, include more than one explicit variable (e.g., flood height and flood velocity). In practice, however, it may be more convenient to express fragility functions in terms of one variable (e.g., flood height) and treat other hazard variables (e.g., flood event duration, associated effects) implicitly. Flood height is a common variable used for fragility analyses, but it is not the only one. Flood velocity may sometimes be more appropriate than flood height for certain failure mechanisms. Depending on the state-of-knowledge available about a given SSC's fragility, it may be appropriate to approximate its fragility function by a step function. For CC-I, the intent of this SR is to allow the analyst to choose a less rigorous approach to develop the fragility function that provides a "representative" estimate of fragility or bounding values. "Representative" estimates may include conservative biases. See XFFR-D1

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-57	CC-II of this requirement differs from CC-I in that it calls for the inclusion of plant-specific information, if available. In some cases, generic information (fragility functions for various structures) is available. Use of generic information is allowable in both CC-I and CC-II, if justified. Technical evaluations may involve engineering judgment. This term is intentionally distinct from "expert judgment" because the use of expert judgment invokes specific requirements. Nonetheless, this requirement specifies that the analyst provide justification that use of engineering judgment is appropriate. See XFFR-D2
XF-N-58	In the evaluation of impacts from coexistent hazards, the first step involves the identification of SSCs that are susceptible to such impacts. For example, a storm surge coincident with high wind loads may cause wind-induced damage of some outside structures, which could lead to propagation pathways for floodwater. Conversely, SSCs inside of robust structures would likely be protected from the effects of high winds. See XFFR-E1
XF-N-59	Ideally, the fragility function of an SSC susceptible to the effects of coexistent hazards would be modified to account for the reduced capacity due to these effects. However, it is recognized that the current state-of-practice may not support a detailed evaluation. Accordingly, this SR does not call for an integrated fragility analysis of coexistent hazards to be performed. Rather, the requirement specifies that coexistent hazards effects on SSCs be addressed, which is intended to provide flexibility to the analyst regarding the manner in which coexistent hazards are considered in the fragility analysis. This could be done, for example, with sensitivity studies. In addition, other sections of this Standard may be consulted when addressing coexistent hazards. See XFFR-E2
XF-N-60	This requirement is intended to identify relevant scenarios for each unscreened flood mechanism. Scenarios include the characteristics of the flood event (e.g., flood height, associated effects, and flood event duration) as well as the effects on the plant (e.g., performance of flood protection, consequences of failures, and resultant plant impacts). Multiple scenarios may be associated with each mechanism (e.g., floods of different severities and timing arising from the same physical phenomena). It is noted that it may be necessary and appropriate to group scenarios for purposes of developing the HLR-XFPR-A. This requirement also addresses mitigation of initiating events associated with the scenarios. It considers mitigation SSCs as well as the features that protect those SSCs through termination or containment of the flood. Examples of relevant protection features that may terminate or contain flood propagation include the following: *(a)* flood barriers; *(b)* sumps (i.e., physical structures that allow for the accumulation and retention of water); *(c)* drains (i.e., physical structures that can function as drains); *(d)* sump pumps, spray shields, and water-tight doors. See XFPR-A1
XF-N-61	This SR requires identification of the characteristics of scenarios associated with ingress of water into the plant. It is the intent of this SR that the characteristics of these scenarios only need to be defined to the extent that they are needed in the analysis. For example, temperature may affect human actions, but if no human actions are credited, then temperature effects may not need to be considered. Similarly, if a room is assumed to be "lost" in the PRA (e.g., SSCs within the room are not credited due to room inundation), the volume of water and duration to room flooding may not need to be considered. See XFPR-A2

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-62	The SR refers to multi-reactor effects in flooding scenarios. This is not intended to refer to a "multi-reactor PRA" but rather to require multi-reactor effects be considered (e.g., when flooding in one reactor may propagate to another reactor due to shared walls or penetrations). See XFPR-A3
XF-N-63	This SR is not applicable to PRAs performed for a single reactor plant. See XFPR-A3, XFPR-C10
XF-N-64	In performing the screening of areas and developing an associated basis, the analyst may consider the requirements of internal flood as providing potentially helpful considerations for development of flood propagation scenarios. However, those SRs are not cross-referenced here to allow flexibility in specifying the associated basis. See XFPR-A4
XF-N-65	This requirement focuses on development of timelines for flood scenarios. Relevant timeline considerations may include the following: *(a)* warning time; *(b)* period of inundation or other defined mission time; *(c)* water surface elevations that are relevant to plant response (e.g., water surface elevations that trigger actions by plant personnel); *(d)* the expected plant mode(s) during the flood event duration; *(e)* necessary human actions, including cues, indications, and notifications. See XFPR-A6
XF-N-66	The initiating flood hazard mechanism may result in actions to re-configure or change the operating mode of the plant prior to the onset of the external flood-induced initiating event or in response to warnings. Therefore, in addition to initiating events caused directly by the flood event (e.g., LOOP, loss of ultimate heat sink availability, loss of functions due to loss of SSCs), this SR requires that "indirect" initiating events be considered (e.g., initiating events caused by actions to re-configure the plant). In addition, coexistent hazards may also lead to direct or indirect initiating events (e.g., due to wind-borne debris or seismically induced failures of SSCs). This SR is not intended to require performance of a "shutdown PRA." The following considerations may be applicable when identifying initiating events as required by this SR: *(a)* human actions-associated plant flood response activities may lead to initiating events; *(b)* a single hazard mechanism may unfold in such a way that failure of SSCs may "map to" multiple initiating events. Given the unique challenges that may arise during a flood event, this SR is also intended to ensure that the analyst considers a range of available information sources related to flood-related challenges to the plant. Relevant flood experience may include events that have occurred at the site as well as industry operating experience (including international experience). In addition, review of operating experience may include the review of situations in which actions were taken in response to warnings (e.g., to install temporary flood protection features) but for which actual plant inundation did not ultimately occur. See XFPR-B1, XFPR-B2
XF-N-67	The internal events PRA model may or may not include event sequences involving two or more reactors or radionuclide sources. However, if external flood-induced failures impact two or more reactors or sources, such event sequences may be introduced for treatment of this hazard group. See XFPR-B1, XFPR-D1
XF-N-68	XFPR-B3 simply requires that initiating events that are identified in XFPR-B1 and XFPR-B2 be included in the external flooding PRA. See XFPR-B3

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-69	The external flooding PRA model can be based on (a) the internal events, at-power, internal flooding, or other PRA model (or combination thereof) adapted to include those aspects that, due to the effects of an external flood, are different from the corresponding aspects of the selected model, (b) models developed specifically to incorporate the external flood plant response, or (c) a combination of (a) and (b). This SR provides requirements for approach (a). In some cases, it may be most appropriate to begin with the low-power, shutdown (LPSD) PRA (if available) as the basis for the External Flooding PRA Plant Response Model (e.g., in cases where a controlled plant shutdown is a fundamental component of the plant flood response strategy). In selected cases involving coexistent hazards, it may be appropriate to start with the PRA model associated with another hazard (e.g., seismic PRA in the case of tsunami hazard assessment). The characteristics of the floods resulting from different mechanisms may be different. For example, a storm surge event may be associated with a high water level and significant dynamic loads but provide substantial warning time to allow a plant to take action. Conversely, a LIP event may be associated with lower water levels and less dynamic flood but fail to provide a warning that would initiate plant response. Moreover, the site flood response to these two mechanisms may differ substantially. For example, the site may utilize an external barrier (e.g., ring levee or flood wall) to protect against a surge event. However, that barrier does not protect against (and may exacerbate) flooding from precipitation events. Instead, the plant may utilize active features (e.g., sump pumps) to protect against precipitation hazards. Due to the different characteristics of the flood events and the divergent plant response, it may be potentially inappropriate (and impractical) to create a single model of plant response that is able to incorporate all response strategies. The peer review for the PRAs used as part of the basis for the External Flooding PRA could include findings that may not yet be resolved. This SR requires that the analyst(s) verify that findings from the peer review of the PRA(s) that form the basis for the external flooding PRA that are relevant to the external flooding PRA be resolved. Findings related to PRAs that are irrelevant to an external flooding PRA (e.g., SGTR, certain RCB breaches) need not be addressed as part of this SR. This SR also requires that the analysts ensure that the disposition of peer review findings does not adversely affect the development of the external flooding PRA. In some cases, a review of the disposition of peer review findings from the internal events PRA may lead to an update of the external flooding PRA plant response model. See XFPR-C1
XF-N-70	XFPR-C3 is intended to capture the potential for SSC failures to be correlated. For example, co-located SSCs may fail concurrently once floodwaters enter the room housing those SSCs (e.g., due to overtopping or failure of the flood barrier protecting the room). In addition, there could be correlation due to factors related to associated effects (e.g., debris loads) that are not related only to spatial co-location. It is also recognized that there are more complex correlation effects (e.g., a single crew may construct sandbag barriers at multiple locations around the site, which may lead to the correlation of the failures of the barriers). However, it is expected that these complex correlation scenarios will be treated using simplified approaches or neglected if the state-of-practice does not support inclusion of the correlation in the model. See XFPR-C3
XF-N-71	See notes for XFPR-C1 See XFPR-C6

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-72	Coexistent hazards may result in plant effects that are different or more severe than those caused only by flooding. For example, an earthquake may cause seismically induced flooding (e.g., due to seismically induced tsunami or dam failures) as well as increased failure potential for SSCs (e.g., damage to flood protection features that are not seismically robust). Another example of coexistent hazards are concurrent wind and surge flooding from a hurricane. In addition to the effects of the surge flooding, the hurricane winds may also lead to rain entering through openings created by damage from high winds (which may be addressed as part of the high winds PRA) and damage to sheet metal credited to protect against wave runup on a building. In addressing coexistent hazards, it is recommended to refer to other parts of this Standard. See XFPR-C7
XF-N-73	The intent of this requirement is to ensure that multi-reactor effects are addressed within the PRA for the reactor under study. For example, this SR is intended to ensure (a) resources credited to the reactor under analysis would be available given that other reactor(s) might compete for the same resources, (b) the external flooding PRA for one reactor captures the effect of failures at the other reactor(s) (e.g., failures of shared flood barriers or other SSCs as well as the potential for flood inundation pathways between reactors), and (c) the effect to on-site accessibility is addressed. This SR is not intended to require a multi-reactor PRA. See XFPR-C10
XF-N-74	Non-light water reactors (non-LWRs) may choose to avoid including large quantities of water near risk-significant structures. Thus, the internal flood PRA model may not be insightful to the XFEL. See XFPR-D1, XFPR-D2
XF-N-75	Many SSCs that would be pertinent to external flooding events may not be identified in the internal events PRA. There is an expectation in this SR to consider barriers, doors, penetrations and seals, sump pumps, and other equipment uniquely related to external flood response. The internal flood PRA may be a helpful resource for identifying SSCs that are relevant to the external flooding PRA. See XFPR-D2
XF-N-76	XFPR-D3 requires augmentation of the XFEL from other sources (e.g., other external flooding PRAs developed for other plants). See XFPR-D3
XF-N-77	XFPR-D4 addresses the potential for some SSCs to become relevant to the external flooding PRA due to impacts from coexistent hazards (e.g., high winds damage to metal-clad structures, wind-borne debris, seismic ground motion). See XFPR-D4
XF-N-78	XFPR-D5 addresses SSC functional failure modes (e.g., loss of pump capacity, spurious operation), which are the result of the mechanistic impact of the external flood upon the SSC (e.g., due to overtopping, foundation effects due to saturated soils, submergence, or failure due to hydrostatic loads). This SR could result in SSC functional failure modes that were not identified in the internal events PRA. See XFPR-D5

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-79	The intent of XFPR-E1 is to identify actions required for plant response to flood events. This may include actions that are in the internal events PRA and internal flood PRA as well as events that are not included in any existing models. This includes planned actions important to plant protection as well as those actions required to respond to an event that may be induced by, or occur during, a flooded plant condition. For some flood hazard mechanisms, many plant flood response actions may not be included in the baseline internal events PRA. Some plant flood response actions may be used to protect non-risk-significant areas of the plant. While treatment of these actions is not required for PRA, their effect on credited actions (e.g., effects of asset protection actions on resource availability and organizational behaviors) is an important consideration as it may impact plant resources devoted to safety relevant actions. See XFPR-E1
XF-N-80	The effect of the external flooding hazard on the control room and ex-control room human actions includes, for example, *(a)* additional workload and stress; *(b)* environment under which personnel are working (e.g., weather, inundated areas, heat, lighting, radiation); *(c)* special fitness issues (e.g., issues associated with physical demands from manual construction of flood barriers); *(d)* effects of external flood on mitigation, required response, timing, and potential for physical harm; *(e)* lack of cue availability; *(f)* staffing and communications; *(g)* training and procedures; *(h)* site accessibility (e.g., external flood effects that may preclude access to or movement around the site); *(i)* flood-specific jobs and training; *(j)* other scenario-specific effects (if applicable). See XFPR-E6
XF-N-81	Certain quantification tools utilize approximations that may cause results to become inaccurate when success branches include basic events with high failure probabilities. In recognition of the possibility of high failure probabilities in conjunction with flood specific actions or SSCs subjected to flood conditions, XFPR-F2 is intended to ensure the analysts consider and address the limitations of computational tools when performing quantification. See XFPR-F2
XF-N-82	See notes for XFFR-B1 See XFPR-G1
XF-N-83	The evaluation of potential physical interactions relies upon the specific findings identified during the walkdown (see XFFR-B4). For example, the failure of a flood protection feature (e.g., flood barrier) can impact other SSCs in several ways (e.g., by allowing water ingress or becoming a source of debris that can impact and damage other SSCs). See XFPR-G2

Table XF-1 Notes Supporting External Flooding PRA Requirements (Cont'd)

Number	Notes
XF-N-84	Examples of items to be documented may include (but are not limited to) the following: *(a)* key findings from investigation(s); *(b)* insights from talk-throughs, tabletop exercises, or simulations; *(c)* external flood timelines and plant response strategies for those flood mechanisms analyzed in the external flooding PRA; *(d)* external flood event and fault trees; *(e)* the specific adaptations made in the internal events PRA model to produce the external flooding PRA model and the basis for those adaptations, or a description of ad hoc models developed specifically for the external flooding PRA; *(f)* those external flood-related influences that affect methods, processes, or assumptions used as well as the identification and quantification of the HFEs in accordance with HLR-XFPR-E; *(g)* the recovery human actions included in the plant response model; *(h)* the preparatory human actions included in the plant response model. See XFPR-H1

XF.2 EXPLANATORY MATERIAL ASSOCIATED WITH EXTERNAL FLOODING PRA

This Standard is anticipated to be used by non-LWR reactor designers and vendors prior to site selection (i.e., at the time of design certification application). The external flooding PRA, as well as high winds, seismic, and other external hazard technical elements and sub-elements, will require a site-specific analysis (e.g., External Flood Hazard Analysis). Prior to site selection, designers may seek to perform an external flooding analysis of their design. The following guidelines are provided to aid designers and vendors in identifying a bounding set of analyses for external flooding PRA and external hazards.

(a) The designer or vendor will select the design-basis external hazards (seismic, tornado, etc.) that generally envelop the potential sites where the plant would be located. For seismic events, seismic response analysis is done for different site conditions to obtain a bounding set of responses.

(b) At the time of the design certification, the fragilities of some SSCs may need to be estimated using generic information and design criteria. The external flooding hazard (in terms of hazard curve and ground response spectra) is chosen to envelop the potential sites. The goal of an external flooding PRA at the design-certification application stage is to identify vulnerabilities and risk insights associated with the design. The PRA at this stage also assures the designer that the plant would meet the associated risk criteria when complete. Even without site-specific information, the external flooding PRA should reveal any unique external flooding-induced event sequences and event sequence families. If the external flooding-induced event sequences do not fit into existing release categories, new release categories are defined for which new mechanistic source terms and radiological consequences are needed. This will facilitate the inclusion of seismically induced event sequences into the Risk Integration.

(c) After the design certification application, a site "A" is chosen and the detailed design of the SSCs completed (or checked) using site-specific information. The hazard curve for the site is used in the quantification along with the plant and design-specific fragilities of SSCs. The external-flooding induced risk is re-evaluated to capture the introduction of a site-specific hazard analysis and site, and design-dependent fragility analysis and Risk Integration accomplished using the same process as was used for the design certification.

(d) When a second site "B" is selected, the designer is expected to verify that the site "A" chosen after the design certification application is suitable at site "B"; any needed modification resulting from site characteristics at B will have to be made. Similarly, the external flooding PRA will be modified to represent the site-specific conditions at "B" and the hazard curve for site "B". The SRs that address site-specific information and conditions when a site has been selected should be applied.

Advanced non-LWRs are generally assumed to be simpler, have fewer systems, and rely more on inherent safety features to perform safety functions passively. In the external hazards elements of the PRA, this simplicity is represented in a simpler internal events PRA model, which would produce a much shorter list of components for a fragility evaluation. However, the technical approach to hazards and fragility analysis are technology inclusive and hence, those requirements are believed to be comparable for a constructed or operating plant at a specific site. For plant designs that are simpler, the complexity of the external hazards PRA is expected to decrease.

XF.2.1 Introduction

Site flooding may result from diverse natural phenomena (e.g., hurricanes, precipitation, and tsunamis) as well as human-made events (e.g., operational releases from dams). The characteristics of the flood events arising from different mechanisms are diverse and are generally characterized by several measures of severity such as flood height or elevation, parameters defining associated effects (e.g., debris loads, sedimentation), and flood event duration (e.g., warning time and period of inundation). Because flood events are temporally and spatially dynamic, these severity measures may differ across a site and throughout the flood event.

Plant flood response strategies may differ depending on site characteristics and the flood mechanisms affecting a site. For example, a plant may utilize external flood barriers (e.g., flood walls or berms) to protect against a coastal flooding event but may rely upon pumps and drainage features to address flooding from local intense precipitation. Moreover, special challenges arise during hurricane events that may concurrently result in surge events as well as heavy precipitation (i.e., external barriers may preclude drainage of rainwater). It is also recognized that some sites were not initially designed to withstand large flood events and may instead rely on mitigation approaches (e.g., portable equipment and non-conventional strategies) to address flooding hazards.

Operating experience has demonstrated the diverse nature of plant flood hazards and associated plant effects. Several examples of this past experience include the impact of Hurricane Andrew on Turkey Point in 1993, the river flooding of Cooper Nuclear Station in1994 and Fort Calhoun Station in 2011, the flooding of the Blayais Nuclear Power Plant resulting from a coastal storm in 1999, the internal flooding at Arkansas Nuclear One in 2013, the precipitation-induced flooding at the St. Lucie Plant in 2014, and the response of several U.S. plants to Hurricane Sandy in 2012. In addition, a number of recent findings at U.S. nuclear power plants are described in Information Notice 2015-01: "Degraded Ability to Mitigate Flooding Events in 2015" [XF-21].

XF.2.2 Conceptual Framework for External Flood Requirements

External flooding PRA is an integrated activity requiring close interactions among specialists from different fields (e.g., hazard analysis, fragility evaluation, human reliability analysis, and systems analysis). Moreover, external flooding PRA is associated with unique challenges arising from the following:

(a) the diverse nature of flooding phenomena affecting a site;

(b) the potential for changes in plant mode or state during the flood event duration;

(c) the importance of considering associated effects in addition to flood elevation, which is the most commonly used measure of flood severity;

(d) the potential for long periods of inundation; and

(e) the diversity of plant response strategies that may be employed, including the use of temporary protection and mitigation measures and strategies that involve significant human actions.

The requirements have been written with cognizance of the diverse nature of flood hazards and associated plant responses as well as the current state of practice and experience related to external flooding PRA for nuclear power plants. The goal of many requirements is to ensure that analysts appropriately consider known characteristics, challenges, and issues associated with flooding hazards as well as the effects on SSCs and plant response. However, in many cases, requirements have been written to afford significant flexibility in how these topics are addressed within the external flooding PRA. As the state of practice evolves and experience base grows, future revisions of this Standard will be updated to incorporate this new knowledge (e.g., more detailed flood mechanism-specific considerations may be added).

Expert and engineering judgment as well as implicit treatment of certain issues may be needed to facilitate the development of the external flooding PRA. For this reason, it is important that the analysts clearly and comprehensively document the development of the external flooding PRA and the technical basis for necessary modeling decisions in a manner that facilitates peer review, PRA applications, and PRA upgrades as the state of practice evolves and the experience base grows.

There are three technical elements in the External Flooding PRA. They are as follows:

(a) The External Flood Hazard Analysis element involves the systematic evaluation of the frequency that a specified parameter or set of parameters representing measures of flood severity (e.g., flood elevation) will be exceeded at a site based on a site-specific probabilistic evaluation. The External Flood Hazard Analysis includes identification and screening of mechanisms as well as a probabilistic flood hazard analysis. The probabilistic flood hazard analysis includes consideration of aleatory variabilities and epistemic uncertainties to calculate the frequency (typically the mean frequency) that a specified parameter or set of parameters representing measures of flood severity will be exceeded at a site.

(b) The External Flood Fragility Analysis element evaluates the fragility of plant SSCs as a function of the severity of the external flood.

(c) The External Flood Plant Response Analysis element develops a plant response model that addresses the initiating events and other failures resulting from the effects of external flood that can lead to radionuclide release. The External Flooding PRA model can be based on (a) the internal events/internal flooding, at-power PRA model adapted to include those aspects that, due to the effects of an external flood, are different from the corresponding aspects of the internal events, at-power model, (b) ad hoc models developed specifically to represent the external flood plant response, or (c) a combination of (a) and (b).

XF.2.3 External Flood-Specific Glossary

Flood hazard: Those hydrometeorological, geoseismic, or structural failure phenomena (or combination thereof) that may produce flooding at or near a nuclear power plant site.

Associated effects: Characteristics of the flood event that are not captured solely by flood elevation (height). Includes factors such as wind waves and runup effects; hydrostatic loading; hydrodynamic loading, including debris and water velocities; effects caused by sediment deposition and erosion; clogging due to debris; concurrent site conditions, including adverse weather conditions; and groundwater ingress.

Flood event duration: Defines the length of time that a flood hazard affects the site. Flood event duration typically begins with conditions being met for entry into a flood procedure or notification of an impending flood and ends when flood waters have receded from the site. It typically includes warning time (if available) and period of inundation and recession.

Flood response SSCs: SSCs that may be used to maintain key safety functions during conditions that might occur during an external flood scenario, including SSCs that are indirectly related to maintenance of key safety functions (e.g., barriers that protect SSCs from floodwaters or other related effects).

Passive flood protection feature: A flood protection feature that does not require the change of state of a component for it to perform as intended. Examples include dikes, berms, sumps, drains, basins, yard drainage systems, walls, floors, structures, penetration seals, and external berms/barriers that are under licensee control.

Power block elevation (for purposes of flood hazard assessment): The as-built elevation of the ground surface in the area of the site power block.

Local intense precipitation (LIP): A locally heavy rainfall event, which is typically defined by specifying three parameters: rainfall depth, rainfall duration, and spatial extent (area). LIP is typically associated with small-scale events over geographic areas on the order of 1- to 10-mi^2 and using an assumption that the rainfall rate is aerially uniform although the rainfall rate (intensity) typically varies over the rainfall event. Although total duration of the rainfall event depends on the scenario and site-specific characteristics (e.g., site drainage, susceptibility to ponding of water), LIP rainfall events are typically associated with a short duration (e.g., 1 to 6 hours) of intense rainfall. These intense rainfall events may be imbedded within longer rainfall events and (depending on site drainage characteristics) may affect a site for longer durations. In the context of this Standard, LIP is defined generically and is not limited to stylized deterministic events, such as the so-called 1-hr, 1-mi^2, Probable Maximum Precipitation (PMP) event.

Walkthrough: Step-by-step consideration of a procedure along with, if possible, visits to relevant locations and demonstration of actions.

XF.2.4 External Flood Hazard Analysis Technical Element

The initiating flood hazard mechanism may result in actions to re-configure or change the operating mode of the plant prior to the onset of the external flood-induced initiating event or in response to warnings. Some flooding events may be associated with little or no warning (e.g., tsunamis, certain precipitation events, and floods resulting from failure of nearby dams) while other flooding events may be associated with substantial warning time (e.g., hurricane events and failure of dams located a substantial distance from a site). Depending on available warning time, sites may take actions to change the operating mode of the plant or re-configure the plant prior to the onset of site flooding (e.g., install flood protection, relocate equipment, or implement mitigation approaches). For the purposes of external flooding PRA, the flooding event is assumed to begin (a) when the site receives notification of a potential flood event and initiates site response (e.g., begins monitoring or taking action) or (b), in the case when no warnings are available, when the flood begins to affect the site (e.g., floodwaters rise to an elevation at which site SSCs are affected). It is noted that external flooding PRA plant response models may need to include actions associated with plant shutdown and/or other re-configurations. Moreover, the initiating events (e.g., flood-induced failure of an SSC) that may lead to a plant transient may occur when the plant is in the re-configured or shutdown state.

Coordination with other parts of this Standard may be needed when addressing coexistent hazards because flooding events may occur as a result of or concurrent with other hazards. For example,

(a) A seismic event may induce a tsunami or cause a failure of an upstream dam leading to a flood at a nuclear power plant site. Moreover, a seismic event may also damage on-site flood protection required to protect against the flood that is induced by the seismic event. In these instances, the external flood event may need to be performed in coordination with insights from the Seismic PRA technical element.

(b) A hurricane event may cause a storm surge event (including waves) but may also result in windborne debris, precipitation, and spray. Moreover, the performance of SSCs may be affected by demands placed on them by the concurrent high winds and storm surge. The frequency of certain initiating events (e.g., LOOP) may also increase due to the combined effects of high wind and surge. As a consequence of the complexity of the hazard the external flood model may need to consider the High Winds PRA technical element. If external flood water enters site buildings, the ensuing flood scenario will likely need to consider the Internal Flood PRA technical element. Hazards that may be induced by another hazard or occur concurrent with another hazard are referred to as coexistent hazards in external flooding PRA. Therefore, when addressing coexistent hazards, it may be necessary for analysts performing the external flooding PRA to interact with other technical elements of this Standard or personnel performing PRAs associated with other hazards to ensure appropriate coverage and treatment of coexistent hazards.

The External Flood Hazard Analysis element addresses the systematic evaluation of the frequency that a specified parameter or set of parameters representing measures of flood severity (e.g., flood elevation) will be exceeded at a site based on a site-specific probabilistic evaluation. The External Flood Hazard Analysis includes identification and screening of mechanisms as well as a probabilistic flood hazard analysis. The probabilistic flood hazard analysis includes identification, characterization, and treatment of uncertainties to quantify the frequency that a specified parameter or set of parameters representing flood severity will be exceeded at a site. This is typically represented by a flood hazard function (which is typically referred to as a flood hazard curve); however, as noted below, this is a simplified hazard characterization. The manner in which uncertainties are addressed in the probabilistic flood hazard analysis will differ depending on whether a Category I or II assessment is being performed.

The External Flood Hazard Analysis involves the following key components:

(a) development of a list of possible flood hazard mechanisms at a site;

(b) screening of the list of flood hazard mechanisms to exclude mechanisms that are not applicable to the site or are otherwise associated with sufficiently low risk to the site;

(c) finalization of the list of mechanisms to include in the probabilistic flood hazard analysis;

(d) probabilistic flood hazard analysis for each mechanism (or combination of mechanisms), including the characterization and treatment of uncertainty;

(e) peer review;

(f) documentation.

When characterizing hazards probabilistically, it is useful to distinguish between aleatory variability and epistemic uncertainty:

(a) Aleatory variability captures natural variability or randomness associated with components or events in the PRA, such as characteristics of specific hazards (rainfall amounts, durations, surge heights, etc.), the time it takes for plant personnel to perform specific tasks, and the conditions (ultimate failure stress, water levels) under which components will fail. Aleatory variability is modeled by discrete or continuous probability distributions. For the probabilistic flood hazard analysis, aleatory variability gives rise to a hazard curve. For the probabilistic flood hazard analysis, modeling of aleatory variability is addressed primarily by the requirements of HLR-XFHA-C.

(b) Epistemic uncertainty results from the lack of information or knowledge. Epistemic uncertainties relate to the degree of belief that the analysts have in the representativeness or validity of the PRA model and in its predictions. It arises when there are multiple technically defensible interpretations of data, models, and methods describing a specific phenomenon or process. For example,

in the context of the probabilistic flood hazard analysis, epistemic uncertainties may arise due to different ways to filter or interpret data, different parameter values, or alternate probability distribution choices. Epistemic uncertainty in the probabilistic flood hazard analysis leads to multiple or alternate hazard curves. For the probabilistic flood hazard analysis, epistemic uncertainty (and calculation of mean hazard) is addressed primarily by the requirements of HLR-XFHA-E.

While PRA analysts are often familiar with working with a single flood hazard curve, it is important to recognize that flooding events are complex with temporally and spatially dynamic characteristics. Generally, during a flood event, flood elevations will increase over time until a peak is reached, and then floodwaters will recede. The temporal characteristics of a flood event (e.g., rate of rise and fall of floodwaters) will depend on the flood mechanism(s) under consideration and the characteristics of the underlying phenomena. In addition to this overall “rise and fall” pattern (typically associated with stillwater elevations), total water elevations may be intermittent in nature. This is due, for example, to the effects of wind waves that cause floodwaters to runup against a structure in a quasi-periodic pattern. Flood events are also associated with effects that are not directly related to flood elevations. These associated effects include factors such as high water velocities, debris loads, sedimentation, and erosion.

In addition, the nature and severity of flood hazard characteristics will vary spatially across a nuclear power plant site such that there are different flood depths and effects at different locations across a site. For example, during a storm surge event, wave effects may occur primarily on seaward sides of structures. As another example, during local intense precipitation (LIP) events (which may have varying intensity, duration, and pattern of precipitation), the depth and duration of water ponding and the relationship of the ponding to incipient building flooding may vary locally depending on the site topography. The current state of practice in numerical modeling of flood hazards typically uses a multi-dimensional representation of flood hazards across a geographic region (e.g., flood inundation maps).

While flood events are complex in nature, it is generally necessary to characterize flood hazards using a set of parameters that can be evaluated in the analysis. Floods may generally be characterized by a set of parameters representing flood severity. Relevant parameters may include the following:

(a) flood water depths or elevations;
(b) parameters related to associated effects;
(c) flood event duration.

Associated effects is a term used to capture factors, other than flood height (elevation) that may adversely affect a site. Associated effects may include factors such as wind waves and runup effects; hydrodynamic loading, including debris; effects caused by sediment deposition and erosion; concurrent site hazards, including adverse weather conditions; and groundwater ingress.

Flood event duration is used to define the length of time that the flood event affects the site. It generally begins with conditions being met for entry into a flood procedure or notification of an impending flood (e.g., a flood forecast, notification of dam failure) and ends when water has receded from the site.

Table XF-2 Notes on External Flood Hazard Assessment

Designator	Note
HLR-XFHA-A	The goal of this High Level Requirement (HLR) is to apply a screening process to determine which external flood hazard mechanisms can pose external flood challenges to the site and therefore require the development a probabilistic flood hazard function (typically referred to as a flood hazard curve) or set of hazard functions to characterize the external flood hazard to the plant. The screening process considers individual flood mechanisms as well as plausible combinations of multiple flooding mechanisms (e.g., events associated with flooding from coastal storm surge and river flooding in estuary environments).
HLR-XFHA-B	This HLR focuses on the output of the probabilistic flood hazard analysis. The probabilistic flood hazard analysis is intended to characterize relevant aspects of the flood hazard to the site. These aspects include, but are not necessarily limited to, peak stillwater elevation and total water level, dynamic effects of wave action (e.g., flood velocities) as they relate to impacted structures, debris generation, concurrent effects of winds (e.g., wind velocity, wind wave/runup effects) and timing elements (e.g., warning time and period of inundation) that characterize the hazard to the site. These measures of flood severity (i.e., flood height, associated effects, and flood event duration) may be different at different locations across the site in any given flood event. This HLR (and associated SRs) recognizes that the flood hazard function will typically represent flood hazard severity using a single measure such as flood height or elevation. However, this HLR (and associated SRs) also recognizes that other factors are important and relevant to the characterization of a flooding hazard. Given the current state of practice, these other factors may need to be treated consistently with their projected risk importance. The intent of this requirement is that these aforementioned factors be addressed with flexibility in how that is done. The approach taken should be commensurate with the estimated importance of challenges to SSCs induced by the associated effects.

Table XF-2 Notes on External Flood Hazard Assessment (Cont'd)

Designator	Note
HLR-XFHA-C	The probabilistic flood hazard analysis is performed for external flood hazard mechanisms determined to be applicable to the site (i.e., mechanisms that were not screened out). The SRs for this HLR allow the for use of existing analyses when they may be judged to meet, or may be adjusted to meet, the intent of the requirements in this Section. If the level of expertise to perform the probabilistic flood hazard analysis exists within the utility, those resources may be used. However, the analyst may opt to consult external experts to support these activities.
HLR-XFHA-D	In developing a new probabilistic flood hazard analysis or using an existing probabilistic flood hazard analysis, the supporting information needs to be appropriately up-to-date and characteristic of the site. This requirement is not intended to require continuous checking for and updating of information throughout the process of performing the probabilistic flood hazard analysis over a reasonable time period. Instead, this requirement is intended to require that analysts use appropriately current information at the time analysis is started, which is often referred to as the "model freeze date" or "model cutoff date." When an existing probabilistic flood hazard analysis is utilized (or in cases where the probabilistic flood hazard analysis is performed over a long time frame), it is expected that analysts will assess whether information included in the existing probabilistic flood hazard analysis remains representative of the site and whether the probabilistic flood hazard analysis should be updated to incorporate new information. For example, recordings of large rainfall events may have a significant effect on estimates of hazard frequencies (and the probabilistic flood hazard analysis should be updated to incorporate the new information) whereas new information related to "average" observations may have a negligible effect. This requirement is intended to allow the use of existing probabilistic flood hazard analysis studies (if available) to fulfill this HLR and the associated SRs with the understanding that, if needed, the studies are updated to incorporate new data, models, and methods (i.e., data, models, and methods that have been developed since the probabilistic flood hazard analysis was previously performed).
HLR-XFHA-E	This HLR is intended to ensure that both aleatory and epistemic uncertainties are considered in the probabilistic flood hazard analysis. Aleatory uncertainty gives rise to (results in) a hazard curve, and epistemic uncertainty results in multiple hazard curves, which may result from use of different modeling assumptions/decisions in the probabilistic flood hazard analysis. In the context of epistemic uncertainties, this HLR is intended to ensure that the analyst *(a)* considers diverse and complementary technical interpretations of data, models, and methods (e.g., via literature reviews); and *(b)* addresses or includes those interpretations within the analysis and considers them in the treatment of uncertainty (e.g., to develop a suite of hazard curves), as appropriate.
HLR-XFHA-F	This HLR requires that a walkdown be performed to support the hazard analysis.
HLR-XFHA-G	No commentary provided.

XF.2.5 External Flood Fragility Analysis Technical Element

The External Flood Fragility Analysis focuses on identifying those SSCs that are susceptible to the effects of external floods and evaluating their failure probability as a function of flood severity. As such, it provides a bridge between the External Flood Plant Response Analysis, which organizes the failure logic of the SSCs credited in the plant to reach and maintain safe and stable conditions during and after an external flood event, and the External Flood Hazard Analysis, which identifies the flood mechanisms of potential concern and quantitatively characterizes the range of flood severity to which these SSCs could be subjected.

Table XF-3 Notes on External Flood Fragility Analysis

Section	Note
HLR-XFFR-A	One of the key elements of the external flood fragility evaluation is the identification of the SSCs that will require a fragility evaluation. Only those SSCs that could contribute to the radionuclide release induced by an external flood event are included within the scope of the fragility evaluation. These SSCs are identified in the XFEL. Identification of relevant SSCs is primarily done as part of the External Flood Plant Response Analysis technical element, complemented by a walkdown of the plant (see HLR-XFFR-B and its SRs).
HLR-XFFR-B	In the External Flood Fragility Analysis technical element, an investigation is essential for the identification of vulnerabilities of SSCs to flood impacts (e.g., hydrostatic failures, undermining of foundations), including those due to flood height, associated effects, and flood event duration. The investigation is also essential for observing the current condition of relevant SSCs (e.g., conditions of flood seals or other barriers). The investigation will also ensure that interactions between SSCs that could contribute to radionuclide release in a flood event are identified (e.g., spatial interactions) and that the plant response model is consistent with the actual configuration of the plant. As such, it also supports the External Flood Plant Response Analysis technical element and its associated requirements.
HLR-XFFR-C	This HLR builds upon the data gathered under HLR-XFFR-B and starts their integration with the plant response PRA model. The severity of external flooding may differ across the site. Therefore, this HLR requires that the fragility evaluation consider the characteristics and severity of flooding (i.e., the flood parameters) at the locations of relevant SSCs. The fragility evaluation involves a characterization of the external flood impacts on the relevant SSCs and the determination of the failure mechanisms (e.g., inundation, undermining of a foundation, and structural failure) that may lead to the associated failure mode in the external flood plant response model (e.g., failure of a pump to start/run due to inundation). The potential interactions identified during the investigation (XFFR-B4) are relevant to the evaluation (e.g., failure of a sump pump may lead to inundation of electrical equipment). The external flood impacts may be different, depending on the flood hazard mechanism considered.
HLR-XFFR-D	Using the information gathered in HLR-XFFR-C, HLR-XFFR-D calls for the quantitative evaluation of fragility functions for the appropriate SSCs of the plant response model. This evaluation is specific to (a) the failure mode(s) of the SSCs under consideration and (b) the flood scenario being analyzed, with its flood mechanism attributes and associated flood parameters at the locations of relevant SSCs.
HLR-XFFR-E	In the context of external flood events, two examples of coexistent hazards are given here to illustrate their potential impacts on the fragility of SSCs: *(a)* A seismic event causes the failure of a dam, leading to external flooding of the nuclear power plant site. Due to the seismic impacts, existing flood protection SSCs may be damaged or fail (e.g., multiple flood doors are unable to perform their function due to seismically induced frame deformation). *(b)* High winds lead to a storm surge, causing external flooding. The fragility of SSCs is increased due to the combined loads from the high winds and loads from the storm surge. These examples show that the fragility of SSCs and the plant response model could be significantly altered by coexistent hazards. Therefore, this HLR requires an assessment of the impact of such hazards.

XF.2.6 External Flood Plant Response Analysis Technical Element

Relative to the plant response analysis requirements contained herein, it is assumed that the External Flood PRA analysis team possesses an (a) internal events (preferably including internal flooding) at-power PRA and (b) a mechanistic source term analysis, developed either before or concurrently with the External Flood PRA. It is assumed that

(a) the internal events PRA is used as the basis for the external flooding PRA systems analysis (if appropriate);

(b) the internal events, at-power PRA (if used) has been peer reviewed and complies with the relevant technical elements of this Standard;

(c) the internal flood PRA (if used) has been peer reviewed and complies with the relevant technical elements of this Standard.

The plant response analysis for external flooding PRA may consist of both adding some flooding-related basic events to the internal events systems model and also "trimming" some aspects of that model that do not apply or can be screened out. It is also acceptable to develop an ad hoc systems model tailored especially to the external flooding PRA situation being modeled, instead of starting with the internal events model and adapting it. For example, an ad hoc model may be appropriate for plants that do not rely on barriers for protection of installed plant equipment and, instead, rely primarily on mitigation of the event through alternate equipment not directly modeled in the PRA (e.g., FLEX components). In such cases, the internal events model may not provide a useful basis for the external flooding PRA. However, while an ad hoc model does not rely on the internal events model, its representation in the PRA should be developed in a rigorous and comprehensive manner. Moreover, it is especially important that the ad hoc model be consistent (to the extent appropriate) with the internal events systems model regarding plant response and the cause-effect relationships of the failures. Further, it is then especially important that a peer review be undertaken that concentrates on these aspects. Whichever approach is used, either adapting the internal events systems model or building an ad hoc systems model, it is important that the systems model include all important failures, including failures caused by the flood and non-flooding failures as well as human errors.

Table XF-4 Notes on External Flood Plant Response Analysis

Designator	Note
HLR-XFPR-A	Commentary related to HLRs is provided in conjunction with the associated SRs.
HLR-XFPR-B	Commentary related to HLRs is provided in conjunction with the associated SRs.
HLR-XFPR-C	Commentary related to HLRs is provided in conjunction with the associated SRs.
HLR-XFPR-D	This requirement relates to the development of the XFEL, which includes SSCs that require development of fragilities. It is noted that the internal events model (including the internal flooding model) may not include all SSCs that are relevant to the external flooding PRA and for which fragilities are needed. For example, passive external flood barriers (e.g., flood walls) may not be included in the internal events PRA but will be included in the XFEL.
HLR-XFPR-E	Commentary related to HLRs is provided in conjunction with the associated SRs.
HLR-XFPR-F	Commentary related to HLRs is provided in conjunction with the associated SRs.
HLR-XFPR-G	Commentary related to HLRs is provided in conjunction with the associated SRs.
HLR-XFPR-H	Commentary related to HLRs is provided in conjunction with the associated SRs.

XF.2.7 Notes on External Flood Peer Review

In addition to the general requirements of Section 6, the peer review team shall have collective knowledge and experience in the areas of external flood hazard assessment, flood fragility assessment, external flood investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable], and external flood plant response assessment. In addition, the peer review team may benefit from having at least one member with knowledge of internal flood PRA.

Implementation of plant response procedures at some plants could result in altering the plant's configuration from "at-power" to another mode representative of a standby or shutdown mode. In reviewing the PRA for such sites, the peer review team would benefit from having at least one member with knowledge and experience in the various operational modes of the type of nuclear plant for which the PRA is being conducted.

XF.2.8 External Flood Resources

The sections that follow provide resources that may be useful to the analysts performing various activities associated with the external flooding PRA. It is emphasized that, due to the current and evolving nature of the state of practice and experience related to external flooding PRAs, there is a great deal of diversity in the approaches described in these resources. Moreover, many of the resources were developed for applications not related to nuclear power plant applications (e.g., some probabilistic flood hazard analysis methods focus exclusively on floods with return periods less than 100 years). For this reason, the resources below are provided for information and, individually, should not be interpreted as providing definitive or authoritative references for meeting the requirements of this Standard.

The goal of many requirements in this technical element is to ensure that analysts appropriately consider known characteristics, challenges, and issues associated with flooding hazards as well as SSCs and plant response. In many cases, requirements have been written to afford significant flexibility in how these topics are addressed within the external flooding PRA. It is also recognized that expert and engineering judgment as well as implicit treatment of certain issues may be needed to facilitate development of the external flooding PRA. In exercising this judgment, the resources provided below may provide useful insights regarding the state of practice. Note that this list is not offered as an exhaustive list. Inclusion of these references does not constitute their endorsement by this Standard.

XF.2.8.1 Hazard Technical Element

Site-Scale Precipitation

Electrical Power Research Institute (EPRI), "Local Precipitation-Frequency Studies Development of 1 Hour/1-Square Mile Precipitation-Frequency Relationships for Two Example Nuclear Power Plant Site," EPRI-TR-3002004400, October 2014

National Oceanic and Atmospheric Administration (NOAA), "Precipitation Frequency Atlas of the United States," (includes 6 volumes), USDOC/NOAA/NWS, (dates vary)

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SRNL-STI-2013-00664, Probabilistic Hazard Assessment for Tornadoes, Straight-line Wind, and Extreme Precipitation at the Savannah River Site, Savannah River National Laboratory, 2013

Watershed Scale Precipitation

Barker, B., M. G. Schaefer, J. Mumford, J., and R. Swain, “Part I: A Monte Carlo Approach to Determine the Variability of PMF Estimates” and “Part II: Monte-Carlo Simulation using Variable Precipitation Volume and Temporal Distribution, Bumping Lake Basin Flood Frequency Estimate,” U.S. Bureau of Reclamation, 1997

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EPRI, EPRI-TR-3002003013, “Riverine Probabilistic Flooding Hazard Analysis Pilot Proof-of-Concept Study for a Nuclear Power Plant,” October 2014

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4.3.14 Other Hazards PRA (O)

This Section presents the technical requirements associated with Other Hazards PRA.

The requirements in this Section are divided into the following technical subelements:

(a) Other Hazards Analysis (OHA);
(b) Other Hazards Fragility Analysis (OFR);
(c) Other Hazards Plant Response Analysis (OPR).

4.3.14.1 Objectives and Technical Requirements for Other Hazards Analysis (OHA)

The objectives of the Other Hazards Analysis ensure that

(a) frequencies of occurrence are determined for the other hazards as a function of intensity on a site-specific basis; hazard assessments are performed for secondary hazards, if necessary; and uncertainties are identified and assessed; and
(b) the Other Hazards Analysis is documented to provide traceability of the work.

Table 4.3.14.1-1 High Level Requirements for Other Hazards Analysis

Designator	Requirement
HLR-OHA-A	The analysis shall include a hazard analysis to support the fragility and hazard plant response analyses.
HLR-OHA-B	The documentation of the Other Hazards Analysis shall provide traceability of the work.

Table 4.3.14.1-2 Supporting Requirements for HLR-OHA-A

The analysis shall include a hazard analysis to support the fragility and hazard plant response analyses. (HLR-OHA-A)

Index No. OHA-A	Capability Category I	Capability Category II
OHA-A1	For the hazard analysis, either: *(a)* IDENTIFY the site at which the reactor being analyzed is located; or *(b)* DESCRIBE a bounding site and JUSTIFY that the bounding site bounds the list of sites in the scope of the PRA. See Note O-N-1	
OHA-A2	COLLECT via investigation(s) current data and information for the site, and the region if applicable, to be used for the hazard analysis. See Note O-N-2, O-N-3, O-N-4	
OHA-A3	USE regional data or generic data with analyst judgment in the hazard analysis. See Note O-N-5	USE plant-specific data for the hazard analysis. INCLUDE, as necessary, regional data, generic information, and/or analyst judgment. See Note O-N-5
OHA-A4	When using generic data, DEMONSTRATE that use of generic data is applicable and conservative. See Note O-N-6	When using generic data, DEMONSTRATE that use of generic data is applicable. See Note O-N-6
OHA-A5	When generating hazard curves, USE a parameter that most accurately represents a measure of the intensity of the hazard. See Note O-N-7	
OHA-A6	CALCULATE a family of hazard curves and DERIVE a mean hazard curve accounting for model and parameter uncertainties.	
OHA-A7	If expert elicitation or another use-of-experts process is used in developing the hazard information, SATISFY the requirements of Section 4.2, Use of Expert Judgment.	
OHA-A8	PERFORM a conservative or bounding assessment of the frequency of any secondary hazard occurrences that are not screened out (refer to Requirement HS-C7) and the magnitude of hazard consequences.	PERFORM a best estimate assessment of the frequency of any secondary hazard occurrences that are not screened out (refer to Requirement HS-C7) and the magnitude of hazard consequences.
OHA-A9	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with Other Hazards Analysis in a manner that supports Requirement OPR-D9.	
OHA-A10	For PRAs performed during the pre-operational stage, IDENTIFY assumptions in lieu of as-built and as-operated design or site details that impact the Other Hazards Analysis. See Note O-N-8	

Table 4.3.14.1-3 Supporting Requirements for HLR-OHA-B

The documentation of the Other Hazards Analysis shall provide traceability of the work. (HLR-OHA-B)

Index No. OHA-B	Capability Category I	Capability Category II
OHA-B1	DOCUMENT the process used in the Other Hazards Analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* bounding site or list of specific sites; *(b)* methodologies used for generating the hazard curves; *(c)* technical interpretations that are the basis for the inputs and results; *(d)* investigation procedures; *(e)* investigation team composition and member qualification; *(f)* investigation observations and conclusions; *(g)* secondary hazard assessment; *(h)* results of the Other Hazards Analysis.	
OHA-B2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement OHA-A9) associated with the Other Hazards Analysis.	
OHA-B3	For PRAs in the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated design and site details associated with the Other Hazards Analysis. See OHA-A10 See Note O-N-8	
OHA-B4	For PRAs conducted on a bounding site, DOCUMENT the basis for the selection of the bounding site characteristics that bound the range of sites for which the plant is designed and the justification of the applicability of the bounding site. See Note O-N-9	

4.3.14.2 Objectives and Technical Requirements for Other Hazards Fragility Analysis (OFR)

The primary objectives for the Other Hazards Fragility Analysis ensure that

(a) hazard-induced failure probabilities are calculated as a function of hazard intensity (viz. fragilities) for the relevant failure mechanisms of the structures, systems, and components (SSCs) identified by the plant response analysis; as-built as-operated plant conditions are represented (e.g., via plant walkdowns), to the extent necessary for the analysis;

(b) the Other Hazards Fragility Analysis is documented to provide traceability of the work.

Table 4.3.14.2-1 High Level Requirements for Other Hazards Fragility Analysis

Designator	Requirement
HLR-OFR-A	The Other Hazards Fragility Analysis shall calculate hazard-induced failure probabilities as a function of hazard intensity (viz. fragilities) for the relevant failure modes of the SSCs whose failure may contribute to other hazard-induced plant risk.
HLR-OFR-B	The documentation of the Other Hazards Fragility Analysis shall provide traceability of the work.

Table 4.3.14.2-2 Supporting Requirements for HLR-OFR-A

The Other Hazards Fragility Analysis shall calculate hazard-induced failure probabilities as a function of hazard intensity (viz. fragilities) for the relevant failure modes of the SSCs whose failure may contribute to other hazard-induced plant risk. (HLR-OFR-A)

Index No. OFR-A	Capability Category I	Capability Category II
OFR-A1	For those failure modes identified in Requirement OPR-B1, IDENTIFY all possible relevant SSC failure modes. See Note O-N-10, O-N-11	For those failure modes identified in Requirement OPR-B1, IDENTIFY relevant and realistic SSC failure modes. See Note O-N-10, O-N-11
OFR-A2	INCLUDE the effects of the secondary hazards in the SSC fragility analysis. See Requirement OHA-A8	
OFR-A3	BASE the Other Hazards Fragility Analysis as a function of the same parameter used to represent the intensity of the hazard. See Note O-N-10, O-N-12	
OFR-A4	ESTIMATE conservative hazard-induced failure probabilities for the failure modes identified in Requirement OFR-A1. JUSTIFY (e.g., through available experience and test data) that use of generic data is applicable and conservative. If expert judgment is used, SATISFY the requirements of Section 4.2, Use of Expert Judgment. See Note O-N-10	CALCULATE realistic hazard-induced failure probabilities for the failure modes identified in Requirement OFR-A1. JUSTIFY (e.g., through available experience and test data) that use of generic data is applicable. See Note O-N-10
OFR-A5	INCLUDE findings from investigation(s) of the design, plant, and/or site in the Other Hazards Fragility Analysis to confirm as-built, as-operated or as-designed, as-intended-to-operate site conditions. See Note O-N-3	
OFR-A6	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with Other Hazards Fragility Analysis in a manner that supports Requirement OPR-D9. See Note O-N-10	
OFR-A7	For PRAs performed during the pre-operational stage, IDENTIFY assumptions in lieu of as-built and as-operated design or site details that impact the Other Hazards Fragility Analysis. See Note O-N-8	

Table 4.3.14.2-3 Supporting Requirements for HLR-OFR-B

The documentation of the Other Hazards Fragility Analysis shall provide traceability of the work. (HLR-OFR-B)

Index No. OFR-B	Capability Category I	Capability Category II
OFR-B1	DOCUMENT the process used in the Other Hazards Fragility Analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* methodologies used to quantify the fragilities of SSCs; *(b)* list of SSC fragility values that includes the method of analysis, the dominant failure mode(s), the sources of information, and the location of each SSC; *(c)* observations and conclusions resulting from investigation(s).	
OFR-B2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement OFR-A6) associated with the Other Hazards Fragility Analysis.	
OFR-B3	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details. See OFR-A7 See Note O-N-8	

4.3.14.3 Objectives and Technical Requirements for Other Hazards Plant Response Analysis (OPR)

The primary objectives for the Other Hazards Plant Response Analysis ensure that

(a) the Other Hazards Plant Response Analysis includes hazard-induced initiating events and other failures and the plant's response to them; risk-significant contributors are identified and understood in the context of the plant design, operation, and maintenance;

(b) event sequences are developed based on the plant configuration, and the initiating events and failures (SSCs and human errors);

(c) hazard-specific challenges to human performance are considered in the human actions credited;

(d) the hazard analysis and the hazard-induced SSC failures are integrated with the systems and event sequence models to quantify the model; analysis limitations and uncertainties are understood;

(e) the Other Hazards Plant Response Analysis is documented to provide traceability of the work.

Table 4.3.14.3-1 High Level Requirements for Other Hazards Plant Response Analysis

Designator	Requirement
HLR-OPR-A	The Other Hazards PRA shall include hazard-induced initiating events that cause risk-significant event sequences and/or risk-significant event sequence families.
HLR-OPR-B	The Other Hazards Plant Response Analysis shall include other hazard-induced SSC failures, non-hazard-induced SSC failures, unavailabilities, human errors, and multi-reactor effects that can affect the frequencies of other hazard-induced event sequence families modeled in the PRA.
HLR-OPR-C	Human actions credited in the Other Hazards PRA shall consider hazard-specific challenges to human performance.
HLR-OPR-D	The analysis to quantify the frequency of each modeled event sequence and event sequence family shall integrate the hazard, hazard-induced SSC failures, and the Other Hazards Plant Response Analysis, including uncertainties.
HLR-OPR-E	The documentation of the Other Hazards Plant Response Analysis shall provide traceability of the work.

Table 4.3.14.3-2 Supporting Requirements for HLR-OPR-A

The Other Hazards PRA shall include hazard-induced initiating events that cause risk-significant event sequences and/or risk-significant event sequence families. (HLR-OPR-A)

Index No. OPR-A	Capability Category I	Capability Category II
OPR-A1	IDENTIFY those hazard-induced initiating events and degraded conditions caused directly by the hazard.	
OPR-A2	Using a systematic process and a review of relevant industry experience, IDENTIFY secondary hazard events resulting from consequential hazards that can themselves induce initiating events or fail SSCs modeled in the hazard PRA.	
OPR-A3	INCLUDE consideration of initiating events represented by industry experience (e.g., through review of plant-specific response to past related hazard events, industry operating experience, and other available related hazard evaluations for nuclear plants).	
OPR-A4	INCLUDE in the plant response model the events identified by Requirements OPR-A1, OPR-A2, and OPR-A3 above that cause risk-significant event sequences and/or risk-significant event progression sequences.	

Table 4.3.14.3-3 Supporting Requirements for HLR-OPR-B

The Other Hazards Plant Response Analysis shall include other hazard-induced SSC failures, non-hazard-induced SSC failures, unavailabilities, human errors, and multi-reactor effects that can affect the frequencies of other hazard-induced event sequence families modeled in the PRA. (HLR-OPR-B)

Index No. OPR-B	Capability Category I	Capability Category II
OPR-B1	IDENTIFY those SSCs, and associated failure modes, required to maintain the plant in operation (as applicable for the analysis of a given hazard) or that may be required to respond to the hazard (including consideration of the identified secondary hazard per Requirement OHA-A8) to prevent a release of radioactive material. See Note O-N-13	
OPR-B2	USE the event sequences and the systems logic model from the internal event PRA models as the basis of the Other Hazards Plant Response Analysis. INCLUDE additional event sequences, as applicable, associated with hazard induced initiating events that are not included in the internal events PRA model such as events that impact two or more reactors or sources of radioactive material. See IE-A16 See Note O-N-14	
OPR-B3	ENSURE that the peer review findings for the Initiating Events Analysis PRA, Internal Fire PRA, Internal Flood PRA, and external hazards PRAs (i.e., Seismic PRA, High Winds PRA, and External Flooding PRA), that are relevant to the results of the other hazards PRA are resolved and incorporated into the development of the Other Hazards Plant Response Analysis. See Note O-N-15	
OPR-B4	INCLUDE hazard-induced failures in the plant response model (including consideration of the identified secondary hazards per Requirement OHA-A8). See Note O-N-16	
OPR-B5	MODEL the correlation of hazard-induced SSC failures, as applicable to a given hazard. JUSTIFY the correlation approach used (e.g., by performing sensitivity studies to assess the contribution to the risk results).	
OPR-B6	APPLY the screening criteria SCR-2 or SCR-3 in Table 1.10-1 when screening out SSC fragility failure modes from the Other Hazards Plant Response Analysis.	
OPR-B7	JUSTIFY the inclusion of beneficial hazard-induced SSC failures.	
OPR-B8	ASSESS the event sequence family frequency of the other hazard-induced event sequences. SATISFY Capability Category I (CC-I) of Requirement SC-A7 for success criteria, except where the requirements are not applicable to the other hazards PRA, to confirm that sustained impacts on plant accessibility and emergency response capability do not invalidate the assumed mission time.	ASSESS the event sequence family frequency of the other hazard-induced event sequences. SATISFY Capability Category II (CC-II) of Requirement SC-A7 for success criteria, except where the requirements are not applicable to the other hazards PRA, to confirm that sustained impacts on plant accessibility and emergency response capability do not invalidate the assumed mission time.

Table 4.3.14.3-3 Supporting Requirements for HLR-OPR-B (Cont'd)

The Other Hazards Plant Response Analysis shall include other hazard-induced SSC failures, non-hazard-induced SSC failures, unavailabilities, human errors, and multi-reactor effects that can affect the frequencies of other hazard-induced event sequence families modeled in the PRA. (HLR-OPR-B)

Index No. OPR-B	Capability Category I	Capability Category II
OPR-B9	If new logic is added to the other hazards PRA (e.g., new system modeling, new or modified event sequences), SATISFY the CC-I SRs of the following High Level Requirements (HLRs), as applicable: *(a)* Event Sequence Analysis per HLR-ES-A and HLR-ES-B; *(b)* Success Criteria Development per HLR-SC-A and HLR-SC-B; *(c)* Systems Analysis per HLR-SY-A and HLR-SY-B; *(d)* Data Analysis per HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D; and *(e)* Human Reliability Analysis (specifically for pre-initiators) per HLR-HR-D.	If new logic is added to the other hazards PRA (e.g., new system modeling, new or modified event sequences), SATISFY the CC-II SRs of the following -HLRs, as applicable: *(a)* Event Sequence Analysis per HLR-ES-A and HLR-ES-B; *(b)* Success Criteria Development per HLR-SC-A and HLR-SC-B; *(c)* Systems Analysis per HLR-SY-A and HLR-SY-B; *(d)* Data Analysis per HLR-DA-A, HLR-DA-B, HLR-DA-C, and HLR-DA-D; and *(e)* Human Reliability Analysis (specifically for pre-initiators) per HLR-HR-D.
OPR-B10	For any hazard-induced internal floods included in the other hazards PRA, SATISFY the CC-I for the SRs of HLR-FLSN-A, and Requirements FLESQ-A1, FLESQ-A2, FLESQ-A3, and FLESQ-A4 for Internal Flood Scenarios Development except where the requirements are not applicable to the other hazards PRA.	For any hazard-induced internal floods included in the other hazards PRA, SATISFY the CC-II SRs of HLR-FLSN-A, and Requirements FLESQ-A1, FLESQ-A2, FLESQ-A3, and FLESQ-A4 for Internal Flood Scenarios Development except where the requirements are not applicable to the other hazards PRA.
OPR-B11	For any hazard-induced internal fires included in the other hazards PRA, SATISFY the CC-I requirements in HLR-FPRM-A and HLR-FPRM-B for the Internal Fire Plant Response Model development except where the requirements are not applicable to the other hazards PRA.	For any hazard-induced internal fires included in the other hazards PRA, SATISFY the CC-II requirements in HLR-FPRM-A and HLR-FPRM-B for the Internal Fire Plant Response Model development except where the requirements are not applicable to the other hazards PRA.
OPR-B12	For any hazard-induced external flooding hazards explicitly retained in the other hazards PRA, SATISFY the requirements of the following at CC-I except where they are not applicable: *(a)* External Flood Hazard Analysis SRs of HLR-XFHA-B; *(b)* External Flood Fragility Analysis SRs of HLR-XFFR-A, HLR-XFFR-B, HLR-XFFR-C, and HLR-XFFR-D, to determine the impact of flooding on SSCs; and *(c)* External Flood Plant Response Analysis SRs of HLR-XFPR-A, HLR-XFPR-B, HLR-XFPR-C, HLR-XFPR-D, and HLR-XFPR-E to determine the plant response because of the flood.	For any hazard-induced external flooding hazards explicitly retained in the other hazards PRA, SATISFY the requirements of the following at CC-II except where they are not applicable: *(a)* External Flood Hazard Analysis SRs of HLR-XFHA-B; *(b)* External Flood Fragility Analysis SRs of HLR-XFFR-A, HLR-XFFR-B, HLR-XFFR-C, and HLR-XFFR-D, to determine the impact of flooding on SSCs; and *(c)* External Flood Plant Response Analysis SRs of HLR-XFPR-A, HLR-XFPR-B, HLR-XFPR-C, HLR-XFPR-D, and HLR-XFPR-E to determine the plant response because of the flood.

Table 4.3.14.3-4 Supporting Requirements for HLR-OPR-C

Human actions credited in the Other Hazards PRA shall consider hazard-specific challenges to human performance. (HLR-OPR-C)

Index No. OPR-C	Capability Category I	Capability Category II
OPR-C1	IDENTIFY the human failure events (including recovery actions) from the internal events PRA that are relevant in the context of the other hazards PRA.	
OPR-C2	If additional human actions are credited in the Other Hazards Plant Response Analysis (i.e., not already included in the PRA system or event sequence models), SATISFY the CC-I SRs of HLR-HR-E, HLR-HR-F, and HLR-HR-H except where the requirements are not applicable.	If additional human actions are credited in the Other Hazards Plant Response Analysis (i.e., not already included in the PRA system or event sequence models), SATISFY the CC-II SRs of HLR-HR-E, HLR-HR-F, and HLR-HR-H except where the requirements are not applicable.
OPR-C3	REVIEW in detail the procedures and sequences of events to confirm that the interpretation of the procedures relevant to actions credited in the hazard PRA is consistent with plant operational and training practices.	REVIEW in detail the procedures and sequences of events with plant operations or training personnel to confirm that the interpretation of the procedures relevant to actions credited in the hazard PRA is consistent with plant operational and training practices.
OPR-C4	INCLUDE human failure events (HFEs) in the hazard PRA plant response model such that the HFEs represent the impact of human failures at the function, system, train, or component level, as appropriate.	
OPR-C5	If screening human error probability (HEP) values are used, ENSURE that the screening values bound the values that would result from a detailed quantification of the HEPs, with special consideration for the context of the hazard events in the hazard PRA.	
OPR-C6	USE screening values in accordance with Requirement OPR-C5 for the HEPs for HFEs included in the hazard PRA model. See Note O-N-17	For each selected hazard event, INCLUDE relevant hazard-related effects in the HEP quantifications using detailed analyses for HFEs that are risk-significant, and screening values (per Requirement OPR-C5) acceptable for HFEs that are non-risk-significant in accordance with the CC-II SRs of HLR-HR-G. Attention is to be given to how the hazard situation alters previous assessments in non-hazard analyses as to the influencing factors and the timing considerations in Requirements HR-G4, HR-G6, and HR-G8 except where they are not applicable. See Note O-N-17
OPR-C7	INCLUDE recovery actions that can restore the functions, systems, or components on an as needed basis to provide a more realistic evaluation of risk-significant event sequences. See Note O-N-18	
OPR-C8	For any recovery actions identified in Requirement OPR-C7, INCLUDE relevant hazard-induced effects, including effects that may preclude a recovery action or alter the manner in which it is accomplished, in accordance with Requirements HR-H2 and HR-H4, except where they are not applicable. See Note O-N-18	

Table 4.3.14.3-5 Supporting Requirements for HLR-OPR-D

The analysis to quantify the frequency of each modeled event sequence and event sequence family shall integrate the hazard, hazard-induced SSC failures, and the Other Hazards Plant Response Analysis model, including uncertainties. (HLR-OPR-D)

Index No. OPR-D	Capability Category I	Capability Category II
OPR-D1	In the quantification of event sequence frequency on a plant-year basis, INTEGRATE the hazard, fragility, and systems analyses in the Other Hazards Plant Response Analysis. See Note O-N-19	
OPR-D2	ADDRESS overestimation of risk due to rare event approximations (e.g., where fragilities approach 1.0). See Note O-N-20	
OPR-D3	ENSURE that the discretization of the hazard curves (or other numerical methods used to incorporate the hazard curve in the integration) is appropriate to demonstrate convergence of event sequence frequencies (e.g., the size and number of bins used to discretize the hazard curve).	
OPR-D4	When quantifying the other hazard event sequence family frequencies, SATISFY the following requirements, except where the requirements are not applicable: *(a)* ESQ-A4, ESQ-A6, ESQ-A7; *(b)* ESQ-B1, ESQ-B2, ESQ-B3, ESQ-B5, ESQ-B6, ESQ-B7, ESQ-B8, ESQ-B9, ESQ-B10; *(c)* ESQ-C1, ESQ-C2, ESQ-C3, ESQ-C4, ESQ-C5, ESQ-C6, ESQ-C7, ESQ-C8, ESQ-C9, ESQ-C10, ESQ-C11, ESQ-C12, ESQ-C13, ESQ-C14, ESQ-C15, ESQ-C16, ESQ-C17; *(d)* ESQ-D1, ESQ-D2, ESQ-D3, ESQ-D5, ESQ-D6, and ESQ-D7.	
OPR-D5	USE the quantification process to ensure that the SSCs screened out (based on the screening defined in Requirement OPR-B6) are consistent with Table 1.10-1.	
OPR-D6	USE the hazard, hazard-induced SSC failure probabilities, and the plant response analysis to generate a point estimate for the event sequence family frequency.	QUANTIFY the mean event sequence family frequency and propagate the parameter uncertainty that results from each input (i.e., hazard, hazard-induced SSC failure probabilities, and the plant response analysis). See Note O-N-21
OPR-D7	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with Other Hazards Plant Response Analysis modeling in a manner that supports Requirement OPR-D9.	
OPR-D8	For PRAs performed during the pre-operational stage, IDENTIFY assumptions in lieu of as-built and as-operated design or site details that impact the Other Hazards Plant Response Analysis. See Note O-N-8	
OPR-D9	SATISFY ESQ-E1 with the additional assumptions in each other hazard technical subelement identified in Requirements OHA-A9 for Other Hazards Analysis, Requirement OFR-A6 for other hazard fragility evaluation, and OPR-D7 for Other Hazards Plant Response Analysis.	

Table 4.3.14.3-6 Supporting Requirements for HLR-OPR-E

The documentation of the Other Hazards Plant Response Analysis shall provide traceability of the work. (HLR-OPR-E)

Index No. OPR-E	Capability Category I	Capability Category II
OPR-E1	DOCUMENT the process used in the Other Hazards Plant Response Analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* identification and disposition of SSCs considered for hazard fragility analysis; *(b)* specific modifications made to the internal events PRA model to produce the other hazards PRA model (including treatment of secondary hazard, as applicable); *(c)* hazard-related influences that affect methods, processes, or assumptions used as well as the identification and quantification of the HFEs/HEPs in accordance with the SRs of HLR-OPR-C; *(d)* justification for the use of the correlation approach used (e.g., bounding 100% fragility correlation) in Requirement OPR-B5; *(e)* screening methodology; *(f)* major outputs of the other hazards PRA, such as mean event sequence family frequencies, uncertainty distributions on event sequence family frequencies, results of sensitivity studies, and risk-significant contributors.	
OPR-E2	DOCUMENT the risk-significant contributors (such as initiating events, event sequences, basic events) to other hazard event sequence family frequencies in the PRA results summary. DESCRIBE risk-significant event sequences or risk-significant contributors.	
OPR-E3	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement OPR-D7) associated with the Other Hazards Plant Response Analysis.	
OPR-E4	For PRAs performed on a bounding site or during the pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated design or site details associated with the other hazard plant response model (PRM). See OPR-D8 See Note O-N-8	
OPR-E5	DOCUMENT limitations in the quantification process that would impact applications.	

4.3.14.4 Peer Review Requirements for Other Hazards PRA

4.3.14.4.1 Purpose

This Section states requirements for peer review of an Other Hazards PRA for all modeled plant operating states and source of radioactive materials consistent within the scope of the PRA.

4.3.14.4.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the subjects of systems engineering, evaluation of the relevant hazard(s), and evaluation of how the hazard(s) could damage the nuclear plant or design SSCs, as applicable to the scope of the review.

4.3.14.4.3 Review of PRA Elements to Confirm the Methodology

4.3.14.4.3.1 Hazard Selection

This Section provides requirements for peer review of the Other Hazards Analysis of the PRA.

4.3.14.4.3.2 Hazard-Caused Initiating Events

The peer review team shall evaluate whether the initiating events postulated to be caused by the hazard are properly identified, the SSCs are properly modeled, and the event sequences are properly quantified.

4.3.14.4.3.3 Fragility Analysis Methods and Data

The peer review team shall evaluate whether the methods and data used in the "fragility" analysis of SSCs are adequate for the purpose and meet the relevant requirements of this Standard.

4.3.14.4.3.4 Investigation

The peer review team shall review the investigation(s) [e.g., walkdown(s), interviews, tabletop review, or computerized walkdown, as applicable] of the plant and its surroundings, as applicable, to ensure the validity of the findings of the analysis in terms of screening, any spatial interactions, and the identification of critical failure mechanisms.

For PRAs on plants prior to operation, if plant walkdown(s) is not possible, the peer review team should review the findings of the following information via interviews and reviews (e.g., tabletop reviews, computerized simulations) with engineering personnel to ensure the validity of the findings of the analysis in terms of screening, any spatial interactions, and the identification of critical failure mechanisms.

4.3.14.4.3.5 Quantification Method

The peer review team shall evaluate whether the quantification method used in the PRA is appropriate and provides all of the results and insights needed for risk-informed decisions. If the analysis contains screening assumptions, or assumptions that the analysis team claims to be demonstrably conservative, the peer review team shall review the validity of these assumptions. The review shall focus on the event sequence and event sequence family frequency estimates and uncertainty bounds and on the risk-significant contributors.

4.3.14.5 References for Other Hazards PRA

The following is a list of publications referenced in this Standard.

[O-1] J. Hickman et al., "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," Report NUREG/CR-2300, American Nuclear Society, Institute of Electrical and Electronic Engineers, and U.S. Nuclear Regulatory Commission, 1983

[O-2] M. McCann, J. Reed, C. Ruger, K. Shiu, T. Teichmann, A. Unione, and R. Youngblood, "Probabilistic Safety Analysis Procedures Guide," Report NUREG/CR-2815, Vol. 2, Brookhaven National Laboratory and U.S. Nuclear Regulatory Commission, 1985

[O-3] International Atomic Energy Agency, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants," IAEA Safety Standards Series No. SSG-3, IAEA, Vienna, 2010

[O-4] International Atomic Energy Agency, "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants," IAEA Safety Standards Series No. SSG-4, IAEA, Vienna, 2010

NONMANDATORY APPENDIX O: NOTES AND EXPLANATORY MATERIAL FOR OTHER HAZARDS PRA

O.1 NOTES ASSOCIATED WITH OTHER HAZARDS PRA

Table O-1 Notes Supporting Other Hazards PRA Requirements

Number	Notes
O-N-1	The description of a bounding site can be the identification of an existing site if that site bounds the hazard for other sites under consideration. See OHA-A1
O-N-2	Although a site-specific and plant-specific hazard analysis is always desirable, it is often acceptable to develop a hazard on some other basis (for example, a regional or even generic basis), provided that the uncertainties introduced are acceptable for the applications contemplated. See OHA-A2
O-N-3	Examples of investigations include, but are not limited to, activities such as tabletop review(s), physical/computerized walkdown(s) of the plant, and/or interview(s) with knowledgeable personnel. These investigation(s) may need to be supported by a review of plant documents, especially in the context of external hazard evaluations. This Supporting Requirement (SR) does not imply that each of these activities need to be performed to meet the SR. Further, if appropriate, one investigation can meet the intent of multiple investigation SRs if the investigation is scoped and structured appropriately. See OHA-A2, OFR-A5
O-N-4	Investigations for this SR may also include a review of industry studies depending on the hazard or hazard group being investigated. See OHA-A2
O-N-5	If no generic data was used, this requirement is satisfied. See OHA-A3
O-N-6	In general, the hazard analysis for a given hazard can be described by a multitude of variables related to the "size" of the event. Often, some of these variables are probabilistically dependent on other variables. However, for simplicity, the hazard function is generally described, albeit imperfectly, in terms of a limited number of variables—typically, one. For example, although a proper characterization of the hazard from a potential chemical explosion from a nearby railroad train carrying chemicals should include blast distance, duration, instantaneous pressure duration, shape of the pressure pulse as a function of frequency, chemical form of the explosive, and so on, the hazard would likely be characterized by only one or two of these parameters in any actual analysis (e.g., overpressure). The other variables that would be needed for a "complete" description of the hazard would typically be considered in the response analysis and fragility analysis, or may represent an irreducible variability in the hazard, or some of each. The output of the hazard analysis is a so-called "hazard curve"— actually, a family of hazard curves accounting for uncertainties—of exceedance frequency versus hazard intensity. The PRA Procedures Guide [O-1] has a useful discussion of the general considerations involved in hazard analysis. See OHA-A4

Table O-1 Notes Supporting Other Hazards PRA Requirements (Cont'd)

Number	Notes
O-N-7	For this technical element, which deals with analysis of an entire category of hazard, the term "hazard" in the singular is used for a single and entire category of similar events, or hazard group, and the hazard group is intended to include all "sizes" of such events within the category. For example, the hazard group for "extreme temperature" includes all extreme temperature conditions, no matter how extreme or how infrequent; the hazard group "transportation events" includes all such events arising from nearby transport modes. Within that hazard group, the hazard group "aircraft impact events" includes crashes of all aircraft, of all sizes, and so on. The set of the requirements in this technical element is concerned with detailed PRA analysis of a hazard group. Even though as written it contemplates the analysis of an entire hazard group, it is not intended to restrict the analyst from analyzing only a subgroup or particular hazard events if the differentiation of the subgroup or hazard event from the remainder of the larger hazard group makes sense, presumably because only the subgroup is important and the remainder can be screened out. These hazards can cover such items as heavy load drops, turbine missiles, volcanic, toxic gas, aircraft impact analysis, etc. See OHA-A5
O-N-8	This SR is not applicable to operating plants. See OHA-A10, OHA-B3, OFR-A7, OFR-B3, OPR-D8, OPR-E4
O-N-9	This SR is not applicable to PRAs performed for a specific site. See OHA-B4
O-N-10	Although a site-specific and plant-specific analysis of the fragilities of SSCs provides the most realistic estimates, for the pre-operational stage PRA, it may be acceptable to develop fragility estimates on some other basis (e.g., generic information), provided that the uncertainties introduced are acceptable for the application. For PRAs performed during the pre-operational stage, decoupling the fragility analysis from the hazard analysis to the extent practical allows for the fragility analysis to bounding a range of sites. It may be necessary to use generic data for this purpose. See OFR-A1, OFR-A3, OFR-A4, OFR-A6
O-N-11	Typical fragility analysis approaches of structures and components may not be directly apply to some hazards. The plant system response analysis may be to model the human action (i.e., type and timing) to prevent and mitigate event sequences. For another hazard (e.g., pipeline event), the fragility analysis may be limited to buildings and exposed equipment. In some hazards (e.g., toxic gas), certain SSC failure mechanisms may in fact be the plant operator errors rather than equipment failure. See OFR-A1
O-N-12	To make the PRA analysis tractable, the fragility is recommended to be expressed as a function of the same variable—related to the "size" of the hazard—of which the hazard curves are functions. This allows the convolution of the hazard curves and fragility curves during the quantification step to be done in a mathematically straightforward way. See OFR-A3
O-N-13	This SR highlights the need to consider SSCs for keeping the plant at-power (e.g., off-site power) as well as SSCs that may be used to safely shut down the plant, as a means to assist in defining applicable hazard initiating event states as well as mitigating SSC failures. It is typically not necessary to explicitly enumerate all the specific SSCs required to maintain the plant at-power. Also, multiple-reactor impacts and dependencies should be considered, as appropriate. See OPR-B1
O-N-14	The internal events PRA model may or may not include event sequences involving two or more reactors or radionuclide sources. However, if other hazard induced failures impact two or more reactors or sources, such event sequences may be introduced for treatment of this hazard group. See OPR-B2
O-N-15	The internal fire PRA, internal flood PRA, and external hazards PRAs may not be applicable to all other hazards and thus may not be relevant to every other hazard PRA. See OPR-B3

Table O-1 Notes Supporting Other Hazards PRA Requirements (Cont'd)

Number	Notes
O-N-16	The analysis may group hazard-induced failures if the leading failure in the group is modeled. The event trees and fault trees from the internal events, at-power PRA model are generally used as the basis for the hazard-initiated event sequences/event trees. This captures the thinking that has gone into their development and assists in allowing comparisons between the internal events PRA and the hazard event PRA to be made on a common basis. See OPR-B4
O-N-17	The human-error probabilities may be increased for some hazard actions, compared to the probabilities assigned in analogous internal events initiated sequences. A typical hazard Human Reliability Analysis aspect is consideration of the possibility that the hazard can cause damage or plant conditions that preclude personnel access to safety equipment or controls, thereby inhibiting human actions that might otherwise be credited. This information is most effectively collected during walkdowns, which must be structured to search for access issues. See OPR-C6
O-N-18	The restoration of safety functions can be inhibited by any of several types of causes, including damage or failure, access problems, confusion, loss of supporting personnel to other post-hazard-recovery functions, and so on. Careful consideration of these causes must be given before recoveries are credited in the initial period after the occurrence of the hazard. This is especially true for externally caused loss of off-site power (LOOP), given that the damage could be to switchyard components or to the off-site grid towers, which are generally difficult to fix quickly. See OPR-C7, OPR-C8
O-N-19	Several documents contain detailed discussions that provide general guidance on how to approach the PRA of a hazard. Examples include the PRA Procedures Guide [O-1], the Probabilistic Safety Assessment Procedures Guide [O-2], and the IAEA Standard Safety Series about the Development and Application of Probabilistic Safety Assessment for Nuclear Power Plants, for Level 1 [O-3] and Level 2 [O-4]. Some of the commentary herein is adapted from these guides. The typical hazard PRA methodology consisting of hazard analysis, fragility analysis, and plant system response analysis and the quantification of risk metrics may have to be modified to suit a particular hazard. When a hazard cannot be screened out using the screening requirements, further analysis is needed, such as a detailed probabilistic hazard analysis to obtain a refined estimate of the frequency of occurrence (e.g., biological events or release of chemicals) or exceedance frequencies (e.g., extreme summer temperature). The systems and event sequence models for an “other” hazard (i.e., internal or external hazard) PRA is commonly based on the internal events, at-power PRA model, to which are added hazard initiating events, SSC failure probability basic events derived from the specific hazard fragility analysis, as well other basic events (e.g., new or adjusted HEPs for the specific hazard). Other factors to be considered include unique aspects of common causes, fragility correlations, any warning time available to take mitigating steps, and the possibility of recovery actions. Internal events event sequence models may also be adjusted, or some sequences not used for a given hazard model. Not using certain parts of the internal events PRA model in the hazard PRA model is common (such adjustments take the form of explicitly deleting the logic in the hazard PRA or by bypassing or directly failing the logic, as appropriate). Depending upon the hazard, new system fault tree logic and/or event sequence logic may need to be developed and added into the PRA model. The frequency of radionuclide release for a hazard model is often (depending upon the hazard) calculated by a convolution of the fragilities and system/sequence models over the relevant range of hazard intensities. The details of the convolution quantification process depend upon the PRA software used. See OPR-D1

Table O-1 Notes Supporting Other Hazards PRA Requirements (Cont'd)

Number	Notes
O-N-20	For some hazards, some SSCs whose hazard-induced failure is important to safety at higher levels will not fail or will fail with only modest probability. The modeling of the non-failure (that is, the “success”) of such SSCs is an important aspect of the plant response model, and excluding these “success” states can lead to erroneous PRA results. Depending upon the specific PRA software used, this success model may or may not include explicit incorporation of “success” basic events into the fault trees and/or event trees; some PRA software quantification algorithms can automatically calculate the complement (success) events without the need for the analyst to explicitly identify and incorporate them into the logic manually. See OPR-D2
O-N-21	This SR is analogous to ESQ-E2 specifying the performance of parametric uncertainty analysis. This SR is not stating that parametric uncertainty analysis is required to be used as the base quantification method for producing the event sequence family results of the other hazard model. See OPR-D6

O.2 EXPLANATORY MATERIAL ASSOCIATED WITH OTHER HAZARDS PRA

The non-LWR PRA Standard is anticipated to be used by non-LWR reactor designers and vendors prior to site selection (i.e., at the time of design-certification application). The other hazards PRA, as well as external flooding, seismic, and high winds technical elements and subelements, will require a site-specific analysis (e.g., Other Hazards Analysis). Prior to site selection, designers may seek to perform another hazards analysis of their design. The following guidelines are provided to aid designers and vendors in identifying a bounding set of analyses for other hazards PRA and external hazards:

(a) The designer or vendor will select the design-basis external hazards (seismic, tornado, etc.) that generally envelop the potential sites where the plant would be located. For seismic events, seismic response analysis is done for different site conditions to obtain a bounding set of responses.

(b) At the time of the design certification, the fragilities of some SSCs may need to be estimated using generic information and design criteria. The other hazard (in terms of hazard curve and ground response spectra) is chosen to envelop the potential sites. The goal of an other hazard PRA at the design-certification application stage is to identify vulnerabilities and risk insights associated with the design. The PRA at this stage also assures the designer that the plant would meet the associated risk criteria when complete. Even without site-specific information, the other hazards PRA should reveal any unique other hazard-induced event sequences and event sequence families that could be efficiently addressed during the design. If the other hazard-induced event sequences do not fit into existing release categories, new release categories are defined for which new mechanistic source terms and radiological consequences are needed. This will facilitate the inclusion of other hazard-induced event sequences into the Risk Integration.

(c) After the design-certification application, a site “A” is chosen and the detailed design of the SSCs completed (or checked) using site-specific information. The hazard curve for the site is used in the quantification along with the plant and -design-specific fragilities of SSCs. The other hazard-induced risk is re-evaluated to capture the introduction of a site-specific hazard analysis, and site- and -design-dependent fragility analysis and Risk Integration accomplished using the same process as was used for the design certification.

(d) When a second site “B” is selected, the designer is expected to verify that the site “A” chosen after the design-certification application is suitable at site “B”; any needed modification resulting from site characteristics at B will have to be made. Similarly, the other hazards PRA will be modified to represent the site-specific conditions at “B” and the hazard curve for site “B”. The SRs that address site-specific information and conditions when a site has been selected should be applied.

Advanced non-LWRs are generally assumed to be simpler, have fewer systems, and rely more on inherent safety features to perform safety functions passively. In the external hazards elements of the PRA, this simplicity is represented in a simpler internal events PRA model that would produce a much shorter list of components for a fragility evaluation. However, the technical approach to hazards and fragility analysis are technology inclusive and hence, those requirements are believed to be comparable for a constructed or operating plant at a specific site. For plant designs that are simpler, the complexity of the external hazards PRA is expected to decrease.

4.3.15 Event Sequence Quantification (ESQ)

This Section presents the technical requirements associated with Event Sequence Quantification.

4.3.15.1 Objectives and Technical Requirements for Event Sequence Quantification

The objectives of the Event Sequence Quantification ensure that

(a) the individual parts of the PRA model of event sequences are integrated to obtain a quantification of event sequence frequencies;

(b) quantification is performed using appropriate models and codes;

(c) functional, physical, and human dependencies are addressed;

(d) quantification supports the determination of risk-significant contributors;

(e) uncertainties in the Event Sequence Quantification are characterized; and

(f) the Event Sequence Quantification is documented to provide traceability of the work.

Table 4.3.15.1-1 High Level Requirements for Event Sequence Quantification

Designator	Requirement
HLR-ESQ-A	The individual modeling items of the PRA shall be integrated to support Event Sequence Quantification which shall quantify the frequency of each modeled event sequence and event sequence family. The integration shall include the event sequences, system models, event progression phenomena, barrier failure modes, data, and human reliability analysis elements, and shall account for functional, physical, and human dependencies and recovery actions.
HLR-ESQ-B	Quantification of the event sequences shall be performed using appropriate models and codes, a truncation level sufficiently low to show convergence, and shall address method-specific limitations and features. Quantification shall also address the breaking of circular logic, identification of mutually exclusive event combinations, use of flag events and modules, and performance of Event Sequence Quantification including the use of system successes.
HLR-ESQ-C	The Event Sequence Quantification shall be done in a manner that all identified functional, physical, and human dependencies are addressed.
HLR-ESQ-D	The Event Sequence Quantification results shall be reviewed, and the risk-significant contributors to the frequency of each risk-significant event sequence and event sequence family shall be identified, such as plant operating states, initiating events, hazard groups, event sequence families, event sequences, basic events [equipment unavailabilities and human failure events (HFEs)], plant damage states, event phenomena, and radionuclide transport barrier failure modes, shall be identified consistent with the scope of the PRA. The results shall be traceable to the inputs and assumptions made in the PRA.
HLR-ESQ-E	Uncertainties in the Event Sequence Quantification results shall be characterized and quantified to the extent practical. Key sources of model uncertainty and assumptions shall be identified, and their potential impact on the results shall be understood. Those sources of uncertainty that are not quantified shall be addressed via sensitivity analysis.
HLR-ESQ-F	The documentation of the Event Sequence Quantification shall provide traceability of the work.

Table 4.3.15.1-2 Supporting Requirements for HLR-ESQ-A

The individual modeling items of the PRA shall be integrated to support Event Sequence Quantification, which shall quantify the frequency of each modeled event sequence and event sequence family. The integration shall include the event sequences, system models, event progression phenomena, barrier failure modes, data, and human reliability analysis elements, and shall account for functional, physical, and human dependencies and recovery actions. (HLR-ESQ-A)

Index No. ESQ-A	Capability Category I	Capability Category II
ESQ-A1	GROUP event sequences into event sequence families according to the criteria in the definition of event sequence family in Section 2.2. JUSTIFY grouping event sequences from different sources of radioactive material and plant operating states into the same event sequence family. See Note ESQ-N-1	
ESQ-A2	INTEGRATE the event sequence delineation, system models, data, and Human Reliability Analysis in the quantification process for each radionuclide source, initiating event group, hazard group, plant operating state, and plant evolution, accounting for system dependencies, to arrive at event sequence family frequencies. INCLUDE as applicable event sequences involving two or more reactors or sources of radioactive material in accordance with Requirement ES-A9. See Note ESQ-N-2	
ESQ-A3	CALCULATE the failure probabilities of different radionuclide transport barrier failure modes contributing to each applicable event sequence family frequency. See Note ESQ-N-2	
ESQ-A4	QUANTIFY the frequencies of the modeled event sequence families. See Note ESQ-N-2, ESQ-N-3	
ESQ-A5	CALCULATE a point estimate of the frequency of each modeled event sequence family. INCLUDE all the modeled radionuclide sources, plant operating states, initiating events, and hazard groups within the scope of the PRA using the point estimate values from the initiating event frequencies, plant operating state probabilities, human error probabilities, and basic event probabilities. See Note ESQ-N-2	QUANTIFY the mean frequency of each modeled event sequence family by propagating the uncertainty distributions of the risk-significant input parameters in such a way that the state-of-knowledge correlation is taken into account unless it can be demonstrated that the effect of the state-of-knowledge correlation is not risk-significant. When the correlation between event probabilities can be demonstrated to be not risk-significant, CALCULATE the mean frequency based on the mean values of the risk-significant input parameters and point estimates for the input parameters that are not risk-significant. INCLUDE all the modeled radionuclide sources, plant operating states, initiating events, and hazard groups within the scope of the PRA. See requirements for quantification of event sequences from different hazard groups, including internal flood PRA, internal fire PRA, seismic PRA, high winds PRA, external flooding PRA , and other hazards PRA. See Note ESQ-N-2
ESQ-A6	SELECT quantification methods that are capable of discriminating the risk-significant contributors to the modeled event sequence family frequencies commensurate with the level of detail in the model. See Note ESQ-N-2	

Table 4.3.15.1-2 Supporting Requirements for HLR-ESQ-A (Cont'd)

The individual modeling items of the PRA shall be integrated to support Event Sequence Quantification, which shall quantify the frequency of each modeled event sequence and event sequence family. The integration shall include the event sequences, system models, event progression phenomena, barrier failure modes, data, and human reliability analysis elements, and shall account for functional, physical, and human dependencies and recovery actions. (HLR-ESQ-A)

Index No. ESQ-A	Capability Category I	Capability Category II
ESQ-A7	INCLUDE recovery actions in the quantification process in applicable event sequence families and cutsets. See Requirements HR-H1, HR-H2, and HR-H5 See Note ESQ-N-2	
ESQ-A8	SELECT parameter values for equipment and operator response consistent with the applicable Capability Category I (CC-I) SRs of HLR-HR-D, HLR-HR-G, HLR-HR-H, HLR-DA-C, and HLR-DA-D, including consideration of the event sequence plant conditions and harsh environments, as appropriate for the level of detail of the analysis.	SELECT parameter values for equipment and operator response consistent with the applicable Capability Category II (CC-II) SRs of HLR-HR-D, HLR-HR-G, HLR-HR-H, HLR-DA-C, and HLR-DA-D, including consideration of the event sequence plant conditions and harsh environments, as appropriate for the level of detail of the analysis.
ESQ-A9	USE conservative parameter estimates to characterize event sequence phenomena.	USE realistic parameter estimates to characterize event sequence phenomena for risk-significant event sequence families. USE conservative or a combination of conservative and realistic estimates for non-risk-significant event sequence families. See Note ESQ-N-2

Table 4.3.15.1-3 Supporting Requirements for HLR-ESQ-B

Quantification of the event sequences shall be performed using appropriate models and codes, a truncation level sufficiently low to show convergence, and shall address method-specific limitations and features. Quantification shall also address the breaking of circular logic, identification of mutually exclusive event combinations, use of flag events and modules, and performance of Event Sequence Quantification including the use of system successes. (HLR-ESQ-B)

Index No. ESQ-B	Capability Category I	Capability Category II
ESQ-B1	PERFORM Event Sequence Quantification by using computer codes that have been demonstrated to generate appropriate results when compared to those from accepted algorithms. IDENTIFY method-specific limitations and features that could impact the results.	
ESQ-B2	TRUNCATE event sequences and associated system models at a sufficiently low cutoff value that dependencies associated with risk-significant cutsets or event sequence families are not eliminated. If cutsets are merged to create a solution (e.g., where system-level cutsets are merged to create sequence level cutsets), then CONFIRM truncation is sufficiently low for the merged cutset solution. See Note ESQ-N-2	
ESQ-B3	ESTABLISH truncation limits by an iterative process of demonstrating that the overall model results converge and that no risk-significant event sequence families are inadvertently eliminated. See Note ESQ-N-2, ESQ-N-4	

Table 4.3.15.1-3 Supporting Requirements for HLR-ESQ-B (Cont'd)

Quantification of the event sequences shall be performed using appropriate models and codes, a truncation level sufficiently low to show convergence, and shall address method-specific limitations and features. Quantification shall also address the breaking of circular logic, identification of mutually exclusive event combinations, use of flag events and modules, and performance of Event Sequence Quantification including the use of system successes. (HLR-ESQ-B)

Index No. ESQ-B	Capability Category I	Capability Category II
ESQ-B4	Where cutsets are used in Event Sequence Quantification, USE the minimal cutset upper bound or an exact solution. JUSTIFY if the rare event approximation is used. See Note ESQ-N-5	
ESQ-B5	Fault tree linking and some other modeling approaches may result in circular logic that must be broken before the model is solved. BREAK the circular logic. When resolving circular logic, DO NOT INTRODUCE unnecessary conservatisms or non-conservatisms. See Note ESQ-N-6	
ESQ-B6	INCLUDE successes of events explicitly modeled in the event sequence logic (e.g., systems, components, barriers) in addition to event failures in the evaluation of event sequences to the extent needed for realistic estimation of event sequence family frequencies. See Note ESQ-N-2, ESQ-N-7	
ESQ-B7	IDENTIFY cutsets (or sequences) containing mutually exclusive events in the results.	
ESQ-B8	CORRECT cutsets containing mutually exclusive events by either *(a)* developing logic to eliminate mutually exclusive situations; or *(b)* deleting cutsets containing mutually exclusive events.	
ESQ-B9	When using logic flags, SET logic flag events to either TRUE or FALSE (instead of setting the event probabilities to 1.0 or 0.0), as appropriate for each event sequence family, prior to the generation of cutsets. See Note ESQ-N-2	
ESQ-B10	If modules, subtrees, or split fractions are used to facilitate the Event Sequence Quantification, USE a process that allows the following: *(a)* identification of shared events; *(b)* correct formation of modules that are truly independent; *(c)* results interpretation based on individual events within modules (e.g., risk significance).	

Table 4.3.15.1-4 Supporting Requirements for HLR-ESQ-C

The Event Sequence Quantification shall be done in a manner that all identified functional, physical, and human dependencies are addressed. (HLR-ESQ-C)

Index No. ESQ-C	Capability Category I	Capability Category II
ESQ-C1	IDENTIFY cutsets with multiple HFEs that potentially impact risk-significant cutset, or event sequence families. See Note ESQ-N-2, ESQ-N-8	
ESQ-C2	ASSESS the degree of dependency among HFEs in the cutset or sequence in accordance with Requirements HR-G6, HR-G7, HR-G8, and HR-G12. See Note ESQ-N-9	
ESQ-C3	When linking event trees, TRANSFER the sequence characteristics (e.g., failed equipment, flag settings) that impact the logic or quantification of the subsequent event development, as well as the sequence frequency. See Note ESQ-N-10	

Table 4.3.15.1-4 Supporting Requirements for HLR-ESQ-C (Cont'd)

The Event Sequence Quantification shall be done in a manner that all identified functional, physical, and human dependencies are addressed. (HLR-ESQ-C)

Index No. ESQ-C	Capability Category I	Capability Category II
ESQ-C4	ASSESS the degree of dependency associated with event phenomena that may impact the capabilities of structures, systems, or components (SSCs) credited to mitigate the event sequences modeled. JUSTIFY any assumptions made that SSCs credited for event mitigation are independent of the relevant event sequence phenomena including harsh environments.	
ESQ-C5	ESTIMATE the challenges (e.g., temperature, pressure loads, debris impingement) to each radionuclide transport barrier in the event sequence modeling using applicable conservative or generic analyses. See Note ESQ-N-11	CALCULATE the challenges (e.g., temperature, pressure loads, debris impingement) to each radionuclide transport barrier in the event sequence modeling using applicable plant- or design-specific analyses for characterizing risk-significant barrier challenges using realistic assumptions. USE conservative or a combination of conservative and realistic treatment for non-risk-significant barrier challenges. If generic calculations are used in support of the assessment, JUSTIFY applicability to the plant being evaluated (e.g., consistent with, or envelope, the plant- or design-specific design features and values). See Note ESQ-N-11
ESQ-C6	INCLUDE model logic necessary to account for event sequence phenomena.	INCLUDE model logic necessary to provide a realistic estimation of the risk-significant event sequence families by accounting for event sequence phenomena. INCLUDE effects of scrubbing on radionuclide release and expected beneficial failures whose omission would distort the quantification results of the risk-significant event sequence families. PROVIDE technical justification (by plant- or design-specific or applicable generic calculations demonstrating the scrubbing mechanisms, or beneficial failures) for the inclusion of any of these features. See Note ESQ-N-2
ESQ-C7	USE conservative estimates for the Human error probabilities (HEPs) of the feasible human actions following the onset of radioactive material release that can impact the human action.	PERFORM detailed analysis for the estimation of HEPs of feasible, risk-significant operator actions following the onset of radioactive material release that can impact the human action consistent with applicable procedures, guidance, or operational philosophy.

Table 4.3.15.1-4 Supporting Requirements for HLR-ESQ-C (Cont'd)

The Event Sequence Quantification shall be done in a manner that all identified functional, physical, and human dependencies are addressed. (HLR-ESQ-C)

Index No. ESQ-C	Capability Category I	Capability Category II
ESQ-C8	DO NOT TAKE CREDIT for equipment survivability or human actions in adverse environments (i.e., beyond equipment qualification limits).	REVIEW risk-significant event sequence families to determine if engineering analyses can support continued equipment operation or operator actions during event sequence progression that could reduce the event sequence frequency. CREDIT continued equipment operation or operator actions when supported by engineering analysis and SATISFY the requirements of SY-A29, HR-H2, ESQ-C2, and ESQ-C4, as applicable. USE conservative or a combination of conservative and realistic analysis for non-risk-significant event sequence families.
ESQ-C9	DO NOT TAKE CREDIT for equipment survivability or human actions in adverse environments (i.e., beyond equipment qualification limits).	JUSTIFY any credit given for equipment survivability or human actions that could be impacted by failure of a radionuclide transport barrier (e.g., based upon an evaluation of environmental conditions allowing human actions and system or documented component environmental qualification). If crediting continued equipment operation or operator actions, SATISFY the requirements of SY-A29, HR-H2, ESQ-C2, and ESQ-C4, as applicable.
ESQ-C10	In the assessment of radionuclide transport barrier capacity, INCLUDE failure mechanisms that lead to gross failure of the radionuclide transport barrier as well as localized failure modes, as applicable, that may result in degraded radionuclide transport barrier performance.	
ESQ-C11	If external hazards are included in the scope of the PRA, INCLUDE barrier failure mechanisms caused by external hazards, such as seismic events, aircraft impact, fires, etc.	
ESQ-C12	For each radionuclide transport barrier included in the event sequence model, IDENTIFY *(a)* the relevant failure modes including gross failures and modes with degraded performance, as well as the associated failure mechanisms, as applicable to each source of radioactive material involved in the radionuclide release; *(b)* relevant physical phenomena that may challenge radionuclide transport barrier integrity; and *(c)* mechanisms associated with any hazard-specific initiating events.	
ESQ-C13	IDENTIFY plant- or design-specific plausible radionuclide transport barrier degradation and failure mechanisms, accounting for unique and reactor-specific design features, and guided by calculating relevant environmental conditions during representative event sequences. JUSTIFY the screening out of any radionuclide transport barrier degradation and failure mechanisms that were identified consistent with SCR-2 or SCR-3 in Table 1.10-1.	

Table 4.3.15.1-4 Supporting Requirements for HLR-ESQ-C (Cont'd)

The Event Sequence Quantification shall be done in a manner that all identified functional, physical, and human dependencies are addressed. (HLR-ESQ-C)

Index No. ESQ-C	Capability Category I	Capability Category II
ESQ-C14	PERFORM a conservative evaluation of the capability of each radionuclide transport barrier to prevent and to mitigate each of the reactor-specific event sequence families as defined in the event sequence logic model and end state assignments.	PERFORM a realistic evaluation of the capability of each radionuclide transport barrier to prevent and to mitigate each of the risk-significant event sequence families as defined in the event sequence logic model and end state assignments. INCLUDE in-service aging effects that may reduce radionuclide transport barrier effectiveness over the reactor lifetime for risk-significant event sequence families. See Note ESQ-N-2
ESQ-C15	If a seismic event or other external hazards are included in the scope of the PRA, ESTIMATE the capacity of each applicable radionuclide transport barrier to withstand the external hazard. See Note ESQ-N-12	If a seismic event or other external hazard is included in the scope of the PRA, CALCULATE the capacity of each applicable radionuclide transport barrier and structure (i.e., develop fragility curves), including interactions with major penetrations, in conformance with the requirements delineated for seismic hazards and other hazards. See Note ESQ-N-12
ESQ-C16	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives in the event sequence quantification dependency analysis in a manner that supports the applicable requirements of HLR-ESQ-E.	
ESQ-C17	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built and as-operated details that influence Event Sequence Quantification dependency analysis. See Note ESQ-N-13, ESQ-N-14	

Table 4.3.15.1-5 Supporting Requirements for HLR-ESQ-D

The Event Sequence Quantification results shall be reviewed, and the risk-significant contributors to the frequency of each risk-significant event sequence and event sequence family shall be identified, such as plant operating states, initiating events, hazard groups, event sequence families, event sequences, basic events (equipment unavailabilities and HFEs), plant damage states, event phenomena, and radionuclide transport barrier failure modes, shall be identified consistent with the scope of the PRA. The results shall be traceable to the inputs and assumptions made in the PRA. (HLR-ESQ-D)

Index No. ESQ-D	Capability Category I	Capability Category II
ESQ-D1	REVIEW a sample of the risk-significant cutsets and event sequence families sufficient to determine that the logic of the cutset or sequence is correct.	
ESQ-D2	REVIEW the results of the PRA for modeling consistency (e.g., event sequence models consistency with systems models and success criteria) and operational consistency (e.g., plant configuration, procedures, and plant- or design-specific and industry experience).	
ESQ-D3	REVIEW results to determine that the flag event settings, mutually exclusive event rules, and recovery rules yield logical results.	
ESQ-D4	For PRAs conducted on reactors with similar plants, COMPARE results to those from similar plants, if available. See Note ESQ-N-15, ESQ-N-16	For PRAs conducted on reactors with similar plants, COMPARE results to those from similar plants, if available, and IDENTIFY causes for differences. See Note ESQ-N-15, ESQ-N-16
ESQ-D5	REVIEW a sampling of non-risk-significant cutsets and event sequence families, including the associated grouped event sequences within those families, to determine they are reasonable and have physical meaning.	
ESQ-D6	IDENTIFY risk-significant contributors to the frequency of each modeled risk-significant event sequence family, including the associated grouped event sequences within those families, using risk significance criteria determined in the SRs of HLR-RI-A, such as initiating events, hazard groups, plant operating states, equipment failures, CCFs, event phenomena, barrier failure modes, and operator errors.	IDENTIFY risk-significant contributors to the frequency of each risk-significant event sequence family, including the associated grouped event sequences within those families, using risk significance criteria determined in the SRs of HLR-RI-A, such as initiating events, hazard groups, plant operating states, equipment failures, CCFs, event phenomena, barrier failure modes, and operator errors. IDENTIFY the contributions of single-reactor and multi-reactor event sequences as applicable. When identifying risk-significant contributors, INCLUDE contributors to the occurrence of both initiating events and event mitigation failures.
ESQ-D7	REVIEW the importance of components and basic events for risk-significant event sequence family frequencies, including the associated grouped event sequences within those families, as determined in the Supporting Requirements (SRs) of HLR-RI-B to ensure that they are consistent with expected results or understand and reconcile the reason for the unexpected results.	
ESQ-D8	PERFORM an assessment to ensure that the cumulative impacts from the initiating events or initiating event groups screened out under Requirement IE-C9 do not affect the risk-significant contributors for the risk assessment. See Note ESQ-N-17	

Table 4.3.15.1-6 Supporting Requirements for HLR-ESQ-E

Uncertainties in the Event Sequence Quantification results shall be characterized and quantified to the extent practical. Key sources of model uncertainty and assumptions shall be identified, and their potential impact on the results shall be understood. Those sources of uncertainty that are not quantified shall be addressed via sensitivity analysis. (HLR-ESQ-E)

Index No. ESQ-E	Capability Category I	Capability Category II
ESQ-E1	ASSESS the effects on event sequence family frequencies of the model uncertainties and related assumptions identified for each technical element by performing a qualitative or quantitative evaluation of the effects of the individual sources of uncertainty or of combinations of interest. See Note ESQ-N-18, ESQ-N-19, ESQ-N-20	
ESQ-E2	CHARACTERIZE the uncertainty distribution of the overall frequency of each modeled event sequence family consistent with the characterization of parameter uncertainties. See Requirements IE-C19, SC-B5, HR-D8, HR-G14, and DA-D3, as applicable. See Note ESQ-N-21	CALCULATE the uncertainty distribution of the overall frequency of each modeled event sequence family by propagating the uncertainty distributions on the parameters for the risk-significant contributors, and those model uncertainties explicitly characterized by a probability distribution, in such a way that the state-of-knowledge correlation between component failure basic event probabilities is accounted for. See Requirements IE-C19, SC-B5, HR-D8, HR-G14, and DA-D3, as applicable. See Note ESQ-N-21

Table 4.3.15.1-7 Supporting Requirements for HLR-ESQ-F

The documentation of the Event Sequence Quantification shall provide traceability of the work. (HLR-ESQ-F)

Index No. ESQ-F	Capability Category I	Capability Category II
ESQ-F1	DOCUMENT the process used in the Event Sequence Quantification, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* records of the process/results when adding nonrecovery terms as part of the final quantification; *(b)* records of the cutset review process; *(c)* a general description of the quantification process including model integration process; treatment of systems or event tree top event successes; the truncation values used; and how recovery and post-initiator HFEs are applied; *(d)* the process and results for establishing the truncation screening values for final Event Sequence Quantification demonstrating that convergence toward a stable result was achieved; *(e)* the total of each plant- or design-specific event sequence family frequency and contributions from the different event sequences, plant operating states, human failure events, common cause failures (CCFs), barrier failure modes, initiating events, hazard groups, and sources of radioactive material, including applicable event sequences associated with releases from two or more reactors or sources of radioactive material, if applicable; *(f)* risk insights associated with the aggregation and disaggregation of contributions from different plant operating states, hazard groups, and sources of radioactive material within the scope of the Event Sequence Quantification; *(g)* the event sequences and their contributing cutsets and the method used to bin the event sequences into intermediate or plant damage states, if used, and event sequence families; *(h)* the approach to the treatment of dependencies across intermediate states or interfaces between modularized event trees, as applicable; *(i)* equipment or human actions that are the key factors in causing the events to not be risk-significant; *(j)* the radionuclide transport barrier failure modes, phenomena, equipment failures, and human actions considered in the development of the Event Sequence Quantification to resolve the release category assignment and characterization of the mechanistic source term; *(k)* the treatment of factors influencing radionuclide transport barrier challenges and capability, as appropriate, for the level of detail of the analysis; *(l)* the basis for the radionuclide transport barrier capacity analysis including the identification of radionuclide transport barrier failure modes and location(s), as appropriate for the type of reactor and its design features; *(m)* the results of the uncertainty and sensitivity analysis (if conducted) including uncertainty distributions for the event sequence frequencies and event sequence family frequencies; *(n)* results of importance analysis; *(o)* a list of mutually exclusive events eliminated from the resulting cutsets and their bases for elimination, as applicable; *(p)* asymmetries in quantitative modeling to provide application users the necessary understanding regarding why such asymmetries are present in the model; *(q)* the process used to illustrate that the computer code(s) used to perform the Event Sequence Quantification will yield the correct results; *(r)* the basis for any parameter estimates not documented to support other PRA elements; *(s)* the approach to ensuring that the use of plant damage states or intermediate end states does not prematurely truncate potentially risk-significant event sequence families and preserves the functional, physical, and human dependencies within the Event Sequence Analysis; *(t)* contributors whose risk significance (or non-risk significance) is driven by assumptions related to scope or level of detail; *(u)* comparison of results to similar plants including causes for risk-significant differences.	

Table 4.3.15.1-7 Supporting Requirements for HLR-ESQ-F (Cont'd)

The documentation of the Event Sequence Quantification shall provide traceability of the work. (HLR-ESQ-F)

Index No. ESQ-F	Capability Category I	Capability Category II
ESQ-F2	DOCUMENT the risk-significant contributors (such as plant operating states, initiating events, event sequences, event sequence families, hazard groups, equipment failures, CCFs, and operator errors) to the frequencies of risk-significant event sequence families. DESCRIBE risk-significant event sequence families in accordance with the definition provided in Section 2.	
ESQ-F3	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirements ESQ-C16, ESQ-E1, and ESQ-E2) associated with the Event Sequence Quantification.	
ESQ-F4	DOCUMENT limitations in the quantification process that would impact applications.	
ESQ-F5	For PRAs conducted in the pre-operational stage, DOCUMENT assumptions and limitations of the Event Sequence Quantification due to the lack of as-built, as-operated details. See ESQ-C17 See Note ESQ-N-14	

4.3.15.2 Peer Review Requirements for Event Sequence Quantification

4.3.15.2.1 Purpose

This Section provides requirements for peer review of the Event Sequence Quantification element of the PRA.

4.3.15.2.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the area of Event Sequence Quantification. The team members assigned to review Event Sequence Quantification shall overlap that assigned to review Initiating Events Analysis and Event Sequence Analysis to ensure consistency between the modeling for these elements. The team members assigned to review the Event Sequence Quantification shall have experience specific to these areas and the capability of recognizing plant and design-specific features of the analyses.

4.3.15.2.3 Review of Event Sequence Quantification to Confirm Results

A review shall be performed on selected event sequences and cutsets. The portion of the event sequences selected for review typically includes the following:

(a) appropriateness of the computer codes used in the quantification;

(b) the truncation values and process including reviewing truncated scenarios for reasonableness;

(c) the recovery analysis;

(d) model asymmetries and sensitivity studies;

(e) the process for generating modules (if used);

(f) logic flags (if used);

(g) the solution of logic loops (if appropriate);

(h) the summary and interpretation of results;

(i) the evaluation of the risk-significant contributors to risk-significant event sequences, event sequence families, and release categories that are determined in the Risk Integration element;

(j) the identification and evaluation of uncertainty in the Event Sequence Quantification.

4.3.15.3 References for Event Sequence Quantification

The following is a list of publications referenced in this Standard.

[ESQ-1] SC-29980-203, "Fluoride-Cooled High Temperature Reactor Licensing Modernization Project Demonstration," Southern Company, September 2019

[ESQ-2] SC-29980-201, "PRISM Reactor Licensing Modernization Project Demonstration," Southern Company, December 2018

[ESQ-3] SC-29980-202, "Westinghouse eVinci™ Micro-Reactor Licensing Modernization Project Demonstration," Southern Company, August 2019

[ESQ-4] SC-29980-200, "High Temperature, Gas-Cooled Pebble Bed Reactor Licensing Modernization Project Demonstration," Southern Company, August 2018

[ESQ-5] NUREG/CR-2728, "Interim Reliability Evaluation Program Procedures Guide," U.S. Nuclear Regulatory Commission, March 3, 1983

[ESQ-6] G. APOSTOLAKIS and S. KAPLAN, "Pitfalls in Risk Calculations," Reliab. Eng., 2, 135, 1981

NONMANDATORY APPENDIX ESQ: NOTES AND EXPLANATORY MATERIAL FOR EVENT SEQUENCE QUANTIFICATION

ESQ.1 NOTES ASSOCIATED WITH EVENT SEQUENCE QUANTIFICATION

Table ESQ-1 Notes Supporting Event Sequence Quantification Requirements

Number	Notes
ESQ-N-1	According to the definition of an event sequence family, the grouping of event sequences ensures that event sequences with similar source, plant operating state, initiating event, plant response, and mechanistic source term will belong to the same event sequence family in a manner consistent with the definition of event sequence family in Section 2.2. This SR requires that similar sequences be grouped into a family, but if there are no similar event sequences, the family may be comprised of a single sequence. The reason for this requirement is that the PRA models may define event sequences at different levels of detail. However, if grouped into families, there will be a better opportunity for consistency across PRAs that model event sequences differently. Aggregation of sequences into families ensures that event sequence screening using SCR-1 is done properly. Event sequence family groupings were exercised during the Licensing Modernization Program's demonstration studies. See [ESQ-1], [ESQ-2], [ESQ-3], and [ESQ-4] See ESQ-A1
ESQ-N-2	Because event sequence families represent groups of event sequences, technical requirements that refer to event sequence families are applicable to all the event sequences within the family. See ESQ-A2, ESQ-A3, ESQ-A4, ESQ-A5, ESQ-A6, ESQ-A7, ESQ-A9, ESQ-B2, ESQ-B3, ESQ-B6, ESQ-B9, ESQ-C1, ESQ-C6, ESQ-C14
ESQ-N-3	The quantification may be accomplished by using either fault tree linking or event trees with conditional split fractions. See ESQ-A4
ESQ-N-4	For example, convergence can be considered sufficient when successive reductions in truncation value of one decade result in decreasing changes in event sequence family frequency, and the final change is less than 5%. See ESQ-B3
ESQ-N-5	The rare event approximation is typically used when basic event probabilities are below 0.1. See ESQ-B4
ESQ-N-6	Guidance for breaking logic loops is provided in NUREG/CR-2728 [ESQ-5]. See ESQ-B5
ESQ-N-7	This accounting may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation that addresses transfers among event trees where the "successes" may not be transferred between event trees. See ESQ-B6
ESQ-N-8	For example, re-quantify the PRA model with human error probability values set to values that are sufficiently high that the risk-significant cutsets are not truncated. See ESQ-C1
ESQ-N-9	For example, explicitly modeling an HFE in an initiating event fault tree, such that the HFE dependency between the at-initiator HFE and post-initiator HFEs can be explicitly addressed using typical human reliability dependency modeling. If the initiating event is a basic event with a hardware and also has a Human Reliability Analysis contribution, use the portion of the initiating event involving operator receiving directions from the control room, and assess this dependency with post-initiator HFEs as part of the human reliability dependency analysis. See ESQ-C2

Table ESQ-1 Notes Supporting Event Sequence Quantification Requirements (Cont'd)

Number	Notes
ESQ-N-10	For example, sequence characteristics can be transferred to another event tree by using the appropriate cutsets. See ESQ-C3
ESQ-N-11	The challenges derived in this requirement will be used in a load-capacity relationship to determine the probability of barrier degradation. See ESQ-C5
ESQ-N-12	For example, bounding estimates may be used to radionuclide transport barrier failure modes and site peak ground accelerations, in conformance with the requirements delineated for seismic hazards and other hazards. See ESQ-C15
ESQ-N-13	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or addressed via the appropriate SRs for uncertainty characterization as the plant transitions from the pre-operational to operational stage. See ESQ-C17
ESQ-N-14	This SR is not applicable to operating plants. See ESQ-C17, ESQ-F5
ESQ-N-15	For example: Why is reactor coolant system boundary (RCB) breach a large contributor for one plant and not another? See ESQ-D4
ESQ-N-16	This SR is likely not applicable to PRAs performed on first-of-a-kind reactors without similar plants to form the basis of the comparison. See ESQ-D4
ESQ-N-17	For PRA applications in which the frequencies and consequences of event sequence families are compared against a frequency-consequence target, cumulative impacts do not exist between event sequence families, and thus, the screening of one event sequence family cannot affect the risk-significant contributors of another event sequence family. See ESQ-D8
ESQ-N-18	For example, possible effects include the introduction of a new basic event, changes to basic event probabilities, and changes in success criteria. See ESQ-E1
ESQ-N-19	The list of assumptions made during model development typically include, as applicable, that specific causes of controlled shutdowns and forced outages starting from at-power conditions are not specified (i.e., the specific causes of exceeding a plant Technical Specification allowed outage time are not quantified separately). See ESQ-E1
ESQ-N-20	The purpose of these requirements is to identify the sources of model uncertainty and assumptions that have not already been addressed in SRs for PRA elements. See ESQ-E1
ESQ-N-21	If a hazard is either not included within the scope of the PRA model or screened out of the PRA model, the uncertainty SR cross-referenced in this SR is not applicable. See ESQ-E2

4.3.16 Mechanistic Source Term Analysis (MS)

This Section presents the technical requirements associated with Mechanistic Source Term Analysis.

4.3.16.1 Objectives and Technical Requirements for Mechanistic Source Term Analysis

The objectives of the Mechanistic Source Term Analysis ensure that

(a) the definition and characterization of release categories is sufficient for the requirements of the Mechanistic Source Term Analysis and Radiological Consequence Analysis;

(b) sources and inventories of radioactive material to be included in the Mechanistic Source Term Analysis within the selected PRA scope are assessed along with the relevant phenomena and barriers that may impact the transport of radioactive material from the radionuclide source to the environment for each modeled event sequence;

(c) mechanistic source terms are calculated;

(d) uncertainties in the mechanistic source terms and transport phenomena are characterized and quantified;

(e) the Mechanistic Source Term Analysis is documented to provide traceability of the work.

Table 4.3.16.1-1 High Level Requirements for Mechanistic Source Term Analysis

Designator	Requirement
HLR-MS-A	The definition and characterization of release categories shall be sufficient for the requirements of the Mechanistic Source Term Analysis and Radiological Consequence Analysis.
HLR-MS-B	The Mechanistic Source Term Analysis shall assess the radionuclide transport barriers and transport mechanisms for each release category.
HLR-MS-C	The mechanistic source term and associated radionuclide transport phenomena shall be calculated.
HLR-MS-D	Uncertainties in the mechanistic source terms and associated radionuclide transport phenomena shall be identified, characterized, and quantified to the extent practical. Key sources of model uncertainty and assumptions shall be identified, and their potential impact on the results shall be understood. Those sources of uncertainty that are not quantified shall be assessed via sensitivity analysis.
HLR-MS-E	The documentation of the Mechanistic Source Term Analysis shall provide traceability of the work.

Table 4.3.16.1-2 Supporting Requirements for HLR-MS-A

The definition and characterization of release categories shall be sufficient for the requirements of the Mechanistic Source Term Analysis and Radiological Consequence Analysis. (HLR-MS-A)

Index No. MS-A	Capability Category I	Capability Category II
MS-A1	ENSURE that the attributes of the radionuclide release categories, defined according to ES-C1, include sufficient characteristics to determine the mechanistic source term. VERIFY that the release category definitions account for the set of physical characteristics defined according to ES-C2, and INCLUDE the following release characteristics as applicable: *(a)* number of reactors involved in the source term or sources of radioactive material released and the initial inventory of each radionuclide group; *(b)* the characteristics of the initiating event (which may affect protective actions); *(c)* quantity of radionuclides released by species in each time phase of release; these quantities may be expressed in terms of inventories and release fractions; *(d)* physical and chemical form of the release for each species including elemental, aerosol, and dust with a specification of aerosol and particle size; *(e)* timing (e.g., time of release including multiple release phases, duration of release(s), warning time before start of release or releases); *(f)* thermal energy of release(s); *(g)* location and elevation of release. See Note MS-N-1	
MS-A2	ENSURE that the set of distinct release categories, defined in ES-C1, is reasonably complete and consistent with the requirements of the Radiological Consequence Analysis.	

Table 4.3.16.1-2 Supporting Requirements for HLR-MS-A (Cont'd)

The definition and characterization of release categories shall be sufficient for the requirements of the Mechanistic Source Term Analysis and Radiological Consequence Analysis. (HLR-MS-A)

Index No. MS-A	Capability Category I	Capability Category II
MS-A3	ENSURE that the release categories defined in ES-C1 are grouped based on the similarity of attributes defined in MS-A1 with differentiation at a level sufficient to support the selected consequence metric in Radiological Consequence Analysis.	ENSURE that the release categories defined in ES-C1 are grouped based on the similarity of attributes defined in MS-A1 with differentiation at a level sufficient to support the selected consequence metric in the Radiological Consequence Analysis and differentiate risk-significant contributors.
MS-A4	JUSTIFY the minimum time period (termination time) for which radiological releases are considered in the characterization of the release category. See Note MS-N-2	
MS-A5	IDENTIFY a bounding sequence to represent each release category. See Note MS-N-3	

Table 4.3.16.1-3 Supporting Requirements for HLR-MS-B

The Mechanistic Source Term Analysis shall assess the radionuclide transport barriers and transport mechanisms for each release category. (HLR-MS-B)

Index No. MS-B	Capability Category I	Capability Category II
MS-B1	ESTIMATE the inventories of each source of radioactive material to support the development of source terms for each modeled release category within the scope of the PRA, including the physical and chemical form of each radionuclide species associated with each source of radioactive material.	CALCULATE the plant- or design-specific inventories of each source of radioactive material to support the development of mechanistic source terms for each modeled release category within the scope of the PRA, including the physical and chemical form of each radionuclide species associated with each source of radioactive material. See Note MS-N-4
MS-B2	IDENTIFY the radionuclide transport barriers from the source to the environment for each modeled source of radioactive material and each modeled release category.	
MS-B3	IDENTIFY the potential failure modes, as applicable, of the radionuclide transport barriers identified in MS-B2.	
MS-B4	For each of the radionuclide transport barriers identified in MS-B2, DETERMINE the radionuclide transport characteristics from each source through the barrier for each release category. See Note MS-N-5	

Table 4.3.16.1-3 Supporting Requirements for HLR-MS-B (Cont'd)

The Mechanistic Source Term Analysis shall assess the radionuclide transport barriers and transport mechanisms for each release category. (HLR-MS-B)

Index No. MS-B	Capability Category I	Capability Category II
MS-B5	ASSESS the following phenomena for inclusion in the Mechanistic Source Term Analysis as applicable: *(a)* heat generation from reactivity additions, decay heat, exothermic chemical reactions, and recriticality; *(b)* heat removal by forced convection, natural convection, conduction, thermal radiation, and endothermic chemical reactions; *(c)* gaseous mass transport by pressure differences (blowdown, expansion, contraction, buoyancy) and by chemical reaction products; *(d)* solid mass transport by diffusion, gravity, steam/water, and other liquids, and liftoff based on radionuclide volatility; *(e)* core and source inventory relocation; *(f)* chemical and physical forms of source term species such as aerosols, particulates/dust, elemental states; *(g)* deposition by gravity, agglomeration, entrainment, plate-out; *(h)* dust and particulate resuspension, and liftoff due to blowdown forces; *(i)* radionuclide decay and buildup from decay of parent nuclides during transport; *(j)* explosions including effects on radionuclide transport barriers and dispersal of radioactive material; *(k)* other transport phenomena that are unique or specific to the reactor type or design. See Note MS-N-6	
MS-B6	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the assessment of radionuclide transport barriers and transport mechanisms for each release category in a manner that supports the applicable requirements of the SRs of HLR-MS-D.	
MS-B7	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built and as-operated details that influence the assessment of radionuclide transport barriers and transport mechanisms for each release category. See Note MS-N-7, MS-N-8	

Table 4.3.16.1-4 Supporting Requirements for HLR-MS-C

The mechanistic source term and associated radionuclide transport phenomena shall be calculated. (HLR-MS-C)

Index No. MS-C	Capability Category I	Capability Category II
MS-C1	For each representative event sequence family identified in MS-A5, USE available and applicable existing or generic source term characteristics defined in Requirement MS-A1.	PERFORM plant- or design-specific analyses to quantify source term characteristics defined in Requirement MS-A1 for each selected representative event sequence family identified in MS-A5 according to requirements delineated in the SRs of HLR-ES-C.
MS-C2	JUSTIFY the applicability of existing or generic source terms selected in Requirement MS-C1, including any post-processing rules or modifications necessary to account for key differences between reference and analyzed plant.	In the calculations performed in Requirement MS-C1, USE appropriate modeling techniques to establish the plant- or design-specific radionuclide release including magnitude and timing by radionuclide isotope group sufficient to define the mechanistic source term, including the treatment of radionuclide transport barriers and transport phenomena identified and assessed in Requirements MS-B2, MS-B3, MS-B4 and MS-B5.

Table 4.3.16.1-4 Supporting Requirements for HLR-MS-C (Cont'd)

The mechanistic source term and associated radionuclide transport phenomena shall be calculated. (HLR-MS-C)

Index No. MS-C	Capability Category I	Capability Category II
MS-C3	USE bounding source terms from applicable existing or generic analyses, including post-processing rules or modifications identified in Requirement MS-C2. See Note MS-N-9	PERFORM plant- or design-specific analysis according to the Supporting Requirements (SRs) of HLR-ES-C with calculations for each release category with the potential to be risk-significant.
MS-C4	JUSTIFY that the treatment of phenomena per Requirements MS-B4 and MS-B5 is sufficient to support Consequence Quantification for the selected metric in the Radiological Consequence Analysis.	JUSTIFY that the treatment of phenomena per Requirements MS-B4 and MS-B5 is sufficient to support Consequence Quantification for the selected metric sufficient to determine risk-significant differences in the consequences of any modeled event sequence family within the scope of the PRA.
MS-C5	USE analysis models, validated computer codes, and data, within known limits of applicability, that have sufficient capability to model or represent the conditions of interest in the determination of mechanistic source terms for each of the reactor-specific release categories and that provide results representative of the plant, including an accepted process for verification and validation of computer programs, or JUSTIFY an alternative method. A qualitative evaluation of a relevant application of codes, models, or analyses that has been used for a similar class of plant (e.g., Owners Group generic studies) may be used. See Note MS-N-10	
MS-C6	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with the calculation of the mechanistic source term and associated radionuclide transport phenomena in a manner that supports the applicable SRs of HLR-MS-D.	
MS-C7	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built and as-operated details that influence the calculation of the mechanistic source term and associated radionuclide transport phenomena. See Note MS-N-7, MS-N-8	

Table 4.3.16.1-5 Supporting Requirements for HLR-MS-D

Uncertainties in the mechanistic source terms and associated radionuclide transport phenomena shall be identified, characterized, and quantified to the extent practical. Key sources of model uncertainty and assumptions shall be identified, and their potential impact on the results shall be understood. Those sources of uncertainty that are not quantified shall be assessed via sensitivity analysis. (HLR-MS-D)

Index No. MS-D	Capability Category I	Capability Category II
MS-D1	IDENTIFY uncertain input parameters influencing source terms for the representative event sequence families in each modeled release category.	
MS-D2	For each release category, CALCULATE a point estimate and CHARACTERIZE the uncertainty for each source term component identified in MS-D1. See Note MS-N-11	For each release category associated with risk-significant event sequence families, *(a)* CALCULATE a mean value for each source term component identified in MS-D1; *(b)* PROVIDE the probabilistic representation of the uncertainty of the parameter estimates; *(c)* if using expert judgment, SATISFY the requirements of Section 4.2, Use of Expert Judgment. For those release categories not associated with risk-significant event sequence families, CALCULATE point estimates, and CHARACTERIZE the uncertainty for each source term component.
MS-D3	ASSESS the consequence effects of the model uncertainties, and related assumptions, identified in MS-B6 and MS-C6, by performing a qualitative or quantitative evaluation of the effects of the individual sources of uncertainty or of combinations of interest.	
MS-D4	CHARACTERIZE the uncertainty distribution of the overall consequence of each modeled release category consistent with the characterization of source term component uncertainties. See Requirement MS-D2.	CALCULATE the uncertainty distribution of the overall consequence of each modeled release category by propagating the uncertainty distributions on the source term components for the risk-significant contributors, and those model uncertainties explicitly characterized by a probability distribution in such as a way that the dependencies between uncertain phenomena are accounted for. See Requirement MS-D2.

Table 4.3.16.1-6 Supporting Requirements for HLR-MS-E

The documentation of the Mechanistic Source Term Analysis shall provide traceability of the work. (HLR-MS-E)

Index No. MS-E	Capability Category I	Capability Category II
MS-E1	DOCUMENT the process used in the Mechanistic Source Term Analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* characterization of each modeled source of radioactive material and inventory; *(b)* technical basis for the adequacy of the definition of each modeled release category; *(c)* assignment of event sequences and event sequence families to each release category; *(d)* relevant radionuclide transport phenomena for each release category; *(e)* models and computer programs used to develop source terms; *(f)* uncertainty and sensitivity analyses for each source term; *(g)* any surrogate risk metrics (e.g., "large release frequency") associated with intermediate states and the relationship of these metrics to the release categories.	

Table 4.3.16.1-6 Supporting Requirements for HLR-MS-E (Cont'd)

The documentation of the Mechanistic Source Term Analysis shall provide traceability of the work. (HLR-MS-E)

Index No. MS-E	Capability Category I	Capability Category II
MS-E2	INCLUDE the quantitative values and assessment of uncertainty for the following parameters of the definition of the source term for each modeled release category: *(a)* number of reactors involved in the source term or sources of radioactive material released and initial inventories; *(b)* quantity of radionuclides released by species in each time phase of release; these quantities may be expressed in terms of inventories and release fractions; *(c)* physical and chemical form of the release for each species including elemental, aerosol, and dust with a specification of aerosol and particle size; *(d)* source term release timing including multiphase releases; *(e)* warning time for evacuation; *(f)* energy of the release; *(g)* elevation of the release.	
MS-E3	DOCUMENT the sources of model and parameter uncertainty, related assumptions, and reasonable alternatives (as identified in MS-B6 and MS-C6 associated with the Mechanistic Source Term Analysis.	
MS-E4	For PRAs performed during pre-operational stage, DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details. See MS-B7 and MS-C7 See Note MS-N-8	

4.3.16.2 Peer Review Requirements for Mechanistic Source Term Analysis

4.3.16.2.1 Purpose

This Section provides requirements for peer review of the Mechanistic Source Term Analysis element of the PRA.

4.3.16.2.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the area of source term analysis. The Mechanistic Source Term Analysis element review shall coordinate, whether through shared team members or other approaches, with the review of the Event Sequence Analysis and Radiological Consequence Analysis elements to ensure consistency between the modeling approaches and desired outputs.

4.3.16.2.3 Review of Systems Analysis to Confirm the Methodology

A review shall be performed of the calculated radionuclide releases (i.e., mechanistic source term), including the following:

(a) the process to group event sequences and/or event sequence families into release categories;
(b) the identification of radionuclide barriers and transport/retention phenomena and the appropriateness of the selected models;
(c) the calculation of the radionuclide releases for the established release categories;
(d) the process to identify and evaluate uncertainties.

4.3.16.3 References for Mechanistic Source Term Analysis

The following is a list of publications referenced in this Standard.

[MS-1] NUREG/CR-0987, SAND79-0299, D. E. BENNETT, "SANDIA-ORIGEN User's Manual," Sandia National Laboratories, 1979

[MS-2] NUREG/BR-0167, "Software Quality Assurance Program and Guidelines," U.S. Nuclear Regulatory Commission, 1993

NONMANDATORY APPENDIX MS: NOTES AND EXPLANATORY MATERIAL FOR MECHANISTIC SOURCE TERM ANALYSIS

MS.1 NOTES ASSOCIATED WITH MECHANISTIC SOURCE TERM ANALYSIS

Table MS-1 Notes Supporting Mechanistic Source Term Analysis Requirements

Number	Notes
MS-N-1	The important characteristics of the initiating event that could impact protective actions include adverse weather conditions that may have caused the initiating event, and external hazard events such as seismic events, external flooding, high winds, and other external hazards that may have adverse impacts on off-site infrastructure needed to implement emergency protective actions such as notification, evacuation, and sheltering. These impacts are covered in the Technical Requirements for external hazards in Seismic PRA, High Winds PRA, External Flooding PRA, and Other Hazards PRA. See MS-A1
MS-N-2	Considerations for justification of the termination time include factors such as the stabilization of releases or the expectation of no significant changes in cumulative release, like revaporization. See MS-A4
MS-N-3	In this context, the bounding event sequence is the one with the highest consequences based on the selected metric. The selection of the bounding consequence event sequence, rather than the average or median consequence, ensures that the release category binning processes do not unintentionally remove event sequences with more severe consequences. This does not imply a conservative analysis of the bounding event sequence, which is examined separately in MS-C3. See MS-A5
MS-N-4	See ORIGEN [MS-1] for a representative calculation technique. See MS-B1
MS-N-5	Examples of radionuclide transport characteristics include the specific release pathway and time-dependent release rates from the fuel, from the reactor coolant system boundary (RCB), and from the reactor building. See MS-B4
MS-N-6	The list of phenomena represents a minimum set of characteristics to be considered in the Mechanistic Source Term Analysis, although the analyst may deem the phenomena unimportant or not applicable based on the required assessment. See MS-B5
MS-N-7	The intent of this SR is to identify all assumptions that the analyst believes can be reasonably closed by the time the reactor is operational. As the plant nears operation, these assumptions should either be closed out with actual as-built and as-operated details or transferred to the standard identification of model uncertainty, the related assumptions, and reasonable alternatives SR as the plant transitions from the pre-operational to operational stage. See MS-B7, MS-C7
MS-N-8	This SR is not applicable to operating plants. See MS-B7, MS-C7, MS-E4
MS-N-9	A bounding source term should be demonstrably conservative for the representative event sequence identified in MS-A5, which was the bounding event sequence within the event sequence families contained within the release category. See MS-C3
MS-N-10	See NUREG/BR-0167 [MS-2] for more information on an accepted process for verification and validation of computer programs. See MS-C5
MS-N-11	This characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the estimate as conservative or bounding. See MS-D2

4.3.17 Radiological Consequence Analysis (RC)

This Section presents the technical requirements associated with Radiological Consequence Analysis. The requirements in this Section are divided into the following technical subelements:

(a) Release Category to Radiological Consequence (RCRE);
(b) Protective Action Parameters and Other Site Data Analysis (RCPA);
(c) Meteorological Data (RCME);
(d) Atmospheric Transport and Dispersion (RCAD);
(e) Dosimetry (RCDO);
(f) Health Effects (RCHE);
(g) Economic Factors (RCEC);
(h) Consequence Quantification (RCQ).

4.3.17.1 Objectives and Technical Requirements for Release Category to Radiological Consequence (RCRE)

The objective of the Release Category to Radiological Consequence subelement ensures that

(a) traceability is provided from the release category Consequence Quantification back to the Mechanistic Source Term Analysis and supporting Event Sequence Analysis;
(b) identification of the factors modeled for the modeled Radiological Consequence Analysis metrics is provided;
(c) the Release Category to Radiological Consequence is documented to provide traceability of the work.

Table 4.3.17.1-1 High Level Requirements for Release Category to Radiological Consequence

Designator	Requirement
HLR-RCRE-A	The Radiological Consequence Analysis shall support the release categories and mechanistic source terms in the Mechanistic Source Term Analysis.
HLR-RCRE-B	The Radiological Consequence Analysis shall identify the degree to which protective actions, meteorology, atmospheric transport, atmospheric dispersion, dosimetry, health effects, and economic factors are modeled for the selected consequence metrics.
HLR-RCRE-C	The documentation of the Release Category to Radiological Consequence transition to Radiological Consequence Analysis shall provide traceability of the work.

Table 4.3.17.1-2 Supporting Requirements for HLR-RCRE-A

The Radiological Consequence Analysis shall support the release categories and mechanistic source terms in the Mechanistic Source Term Analysis. (HLR-RCRE-A)

Index No. RCRE-A	Capability Category I	Capability Category II
RCRE-A1	For the Radiological Consequence Analysis, either *(a)* IDENTIFY the site at which the reactor being analyzed is located; or *(b)* DESCRIBE a bounding site and JUSTIFY that the bounding site bounds the list of sites in the scope of the PRA. See Note RC-N-1	
RCRE-A2	IDENTIFY inputs required for the off-site Radiological Consequence Analysis methodology. At a minimum, INCLUDE the following characteristics for each release category, if applicable: *(a)* the number of plumes; *(b)* the release fraction of each radionuclide group; *(c)* the radionuclide isotopes important to dose or health effects (consistent with the selected health effects in RCRE-B2); *(d)* the release timing and duration of release; *(e)* the warning time for protective actions and hazards impacting protective actions (consistent with the requirements of Protective Action Parameters and Other Site Data Analysis); *(f)* the energy of release; *(g)* the release height; *(h)* the released particle size; *(i)* uncertainties associated with the release.	
RCRE-A3	REVIEW the release category definitions in ES-C1 and the mechanistic source term parameters in HLR-MS-A for the identification of the inputs in RCRE-A2.	

Table 4.3.17.1-3 Supporting Requirements for HLR-RCRE-B

The Radiological Consequence Analysis shall identify the degree to which protective actions, meteorology, atmospheric transport, atmospheric dispersion, dosimetry, health effects, and economic factors are modeled for the selected consequence metrics. (HLR-RCRE-B)

Index No. RCRE-B	Capability Category I	Capability Category II
RCRE-B1	USE the consequence measure(s) selected for the intended application of the PRA per RI-A1.	
RCRE-B2	IDENTIFY the characteristics of the Radiological Consequence Analysis, including the degree to which the following aspects will be evaluated: *(a)* protective actions and other site data; *(b)* meteorology; *(c)* atmospheric dispersion and Transport; *(d)* dosimetry; *(e)* health effects; and *(f)* economic factors.	

Table 4.3.17.1-4 Supporting Requirements for HLR-RCRE-C

The documentation of the Release Category to Radiological Consequence transition to Radiological Consequence Analysis shall provide traceability of the work. (HLR-RCRE-C)

Index No. RCRE-C	Capability Category I	Capability Category II
RCRE-C1	DOCUMENT the process used in the release category to Radiological Consequence Analysis transition, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the inputs/characteristics of Radiological Consequence Analysis identified to be evaluated in RCRE-A2; *(b)* the selected consequences metrics considered.	

4.3.17.2 Objectives and Technical Requirements for Protective Action Parameters and Other Site Data Analysis (RCPA)

The objectives of the Protective Action Parameters and Other Site Data Analysis subelement ensure that

(a) protective actions are properly defined to enable calculation of impact of mitigation strategies in the consequence analysis;

(b) other site, local, and regional data are properly defined and developed to support the consequence analysis;

(c) the Protective Action Parameters and Other Site Data Analysis is documented to provide traceability of the work.

Table 4.3.17.2-1 High Level Requirements for Protective Action Parameters and Other Site Data Analysis

Designator	Requirement
HLR-RCPA-A	Appropriate short- and long-term protective actions shall be used in the modeling.
HLR-RCPA-B	Appropriate site, local and regional population, land use, and geographic data shall be used.
HLR-RCPA-C	The documentation of the Protective Action Parameters and Other Site Data Analysis shall provide traceability of the work.

Table 4.3.17.2-2 Supporting Requirements for HLR-RCPA-A

Appropriate short- and long-term protective actions shall be used in the modeling. (HLR-RCPA-A)

Index No. RCPA-A	Capability Category I	Capability Category II
RCPA-A1	INCLUDE the following short- and long-term protective actions in the model as applicable: *(a)* evacuation; *(b)* sheltering; *(c)* relocation; *(d)* land interdiction/remediation; and *(e)* food interdiction/remediation.	
RCPA-A2	MODEL protective-actions using criteria appropriate to the phase of the incident, including consideration of the following: *(a)* early phase—the first hours or days of an event (sometimes called the emergency phase) when evacuation and sheltering decisions are made and implemented based on facility status and anticipated or in-progress releases; *(b)* intermediate phase—the first weeks to months following a release when protective actions are mainly based on environmental measurements; and *(c)* late/long-term phase—the subsequent months to years following a release when recovery/remediation actions are conducted and completed, and land is released for unrestricted use or condemned.	
RCPA-A3	MODEL protective-actions (e.g., evacuation time estimate, dose criteria for evacuation, sheltering, food and land interdiction) using current applicable documents (e.g., emergency plan, evacuation time estimate study) and recommendation documents from recognized organizations (e.g., Environmental Protection Agency, Food and Drug Administration, state or local bodies, utility). JUSTIFY the use of these applicable documents (e.g., local requirements are more stringent than national requirements, use of international standards in lieu of U.S. standards).	
RCPA-A4	USE one cohort (e.g., for those not complying with protective actions).	USE two or more cohorts in the protective-action modeling (e.g., one cohort for those not complying with protective actions and another cohort for those complying).
RCPA-A5	IDENTIFY assumptions regarding compliance with protective actions (e.g., a uniform percentage of the population is assumed to not evacuate) based on generic data sources. See Note RC-N-2	
RCPA-A6	CREDIT temporary shelter-in-place for the cohort(s) that evacuates, if appropriate for the release category and conditions.	
RCPA-A7	USE protection parameters (e.g., sheltering, shielding values) from generic data sources. See Note RC-N-2	
RCPA-A8	USE simplified evacuation modeling for applicable cohort(s), such as *(a)* radial evacuation; and *(b)* evacuation of full plume exposure pathway emergency planning zone (EPZ).	
RCPA-A9	ESTIMATE the delay time to the start of shelter-in-place and evacuation movement by the general public for applicable cohort(s) INCLUDE the following: *(a)* time of the general emergency declaration by the site per the site emergency procedures (e.g., emergency action level scheme); *(b)* time required for the site to notify off-site public emergency planning officials; *(c)* time required for public officials to initiate notifications to the general public; *(d)* time required for the public to receive specific instructions (e.g., shelter-in-place, evacuate); *(e)* time required to secure personal property; and *(f)* time required to load vehicles for evacuation.	

Table 4.3.17.2-2 Supporting Requirements for HLR-RCPA-A (Cont'd)

Appropriate short- and long-term protective actions shall be used in the modeling. (HLR-RCPA-A)

Index No. RCPA-A	Capability Category I	Capability Category II
RCPA-A10	ESTIMATE the evacuation speed based on evacuation studies specific to the site. A constant average evacuation speed for applicable cohort(s) may be used. ENSURE the speed estimates, as a minimum, incorporate specific consideration of the following: *(a)* daytime versus nighttime impacts; *(b)* adverse weather conditions specific to the site; *(c)* special events (e.g., festivals) that impact traffic conditions by over a factor of two; and *(d)* transient populations.	
RCPA-A11	ADJUST the evacuation speed (e.g., the availability of evacuation routes), delay times, and potential for shelter in place based on the effects of the associated hazard group.	
RCPA-A12	IDENTIFY and CHARACTERIZE parameter uncertainties associated with the protective actions in a manner that supports the applicable requirements of RCQ-C2.	
RCPA-A13	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with protective action in a manner that supports the applicable requirements of HLR-RCQ-C.	
RCPA-A14	For PRAs performed using a bounding site, IDENTIFY assumptions made due to lack of site details that influence protective actions. See Note RC-N-3	

Table 4.3.17.2-3 Supporting Requirements for HLR-RCPA-B

Appropriate site, local and regional population, land use, and geographic data shall be used. (HLR-RCPA-B)

Index No. RCPA-B	Capability Category I	Capability Category II
RCPA-B1	ASSUME local and regional population distributions. JUSTIFY the population distribution assumptions (e.g., population distribution considered bounding for the analysis). See Note RC-N-4	DEVELOP local and regional population estimates based upon recognized demographic sources (e.g., U.S. census data) specific to the site. INCLUDE transient populations (e.g., employees, recreational individuals) in local data. ADJUST data as needed to account for the time period of interest (e.g., projections to a specific year). See Note RC-N-4
RCPA-B2	BASE land use data (e.g., area that is land versus water, fraction of land devoted to farming, agricultural production) on generic sources or simplified assumptions (e.g., all area is habitable land).	BASE land use data (e.g., area that is land versus water, fraction of land devoted to farming, agricultural production) on regional specific sources (e.g., county data, maps). ADJUST the data to incorporate intra-regional differences (e.g., differences between counties within a region).
RCPA-B3	ESTIMATE physical plant characteristics (e.g., building dimensions, stack heights).	USE physical plant characteristics (e.g., building dimensions, stack heights).
RCPA-B4	IDENTIFY the release-source geographic location (e.g., reactor building, mid-way between multiple reactors, longitude/latitude).	

Table 4.3.17.2-3 Supporting Requirements for HLR-RCPA-B (Cont'd)

Appropriate site, local and regional population, land use, and geographic data shall be used. (HLR-RCPA-B)

Index No. RCPA-B	Capability Category I	Capability Category II
RCPA-B5	For PRAs using a bounding site, JUSTIFY the assumptions made in identifying the release source geographic location. See Note RC-N-5	
RCPA-B6	IDENTIFY and CHARACTERIZE parameter uncertainties and related assumptions associated with the protective action parameters and other site data in a manner that supports the applicable requirements of RCQ-C2.	
RCPA-B7	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with protective action parameters and other site data in a manner that supports the applicable SRs of HLR-RCQ-C.	
RCPA-B8	For PRAs performed on a bounding site, IDENTIFY assumptions made due to the lack of site details that influence protective action parameters and other site data. See Note RC-N-5	

Table 4.3.17.2-4 Supporting Requirements for HLR-RCPA-C

The documentation of the Protective Action Parameters and Other Site Data Analysis shall provide traceability of the work. (HLR-RCPA-C)

Index No. RCPA-C	Capability Category I	Capability Category II
RCPA-C1	DOCUMENT the process used in the modeling of protective-actions and other site data, modeling, and parameters, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* protective actions modeled (e.g., shelter-in-place, radial evacuation); *(b)* protective action parameters and bases (e.g., evacuation speed); *(c)* incident phases modeled; *(d)* population distribution and bases; *(e)* land-use data; *(f)* plant physical characteristics (e.g., dimensions, geographic location); *(g)* parameter uncertainty characterization; *(h)* references to generic sources/documents.	
RCPA-C2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in RCPA-A13 and RCPA-B7) associated with the protective-action parameters and other site data.	
RCPA-C3	For PRAs conducted on bounding sites, DOCUMENT assumptions and limitations due to the lack of site details associated with the protective-action parameters and other site data. See RCPA-A14 and RCPA-B8 See Note RC-N-5	

4.3.17.3 Objectives and Technical Requirements for Meteorological Data (RCME)

The objectives of the Meteorological Data subelement ensure that

(a) sources of valid and representative meteorological data are located and used; and

(b) the Meteorological Data analysis is documented to provide traceability of the work.

Table 4.3.17.3-1 High Level Requirements for Meteorological Data

Designator	Requirement
HLR-RCME-A	Accurate meteorological data from spatially representative location(s) shall be compiled.
HLR-RCME-B	The documentation of the Meteorological Data shall provide traceability of the work.

Table 4.3.17.3-2 Supporting Requirements for HLR-RCME-A

Accurate Meteorological Data from spatially representative location(s) shall be compiled. (HLR-RCME-A)

Index No. RCME-A	Capability Category I	Capability Category II
RCME-A1	COMPILE meteorological data records from the site. JUSTIFY that the data are spatially representative of the site (i.e., source) location and the region See Note RC-N-6, RC-N-7	
RCME-A2	SELECT hourly meteorological data for a representative 1-yr period from a location representative of the source and its surroundings.	EVALUATE hourly meteorological data for multiple years from the site location to select a 1-yr period of data that is representative of current conditions.
RCME-A3	COMPILE hourly meteorological data from a location representative of the source and its surroundings. ENSURE meteorological data that does not have large blocks (e.g., weeks) of missing data. JUSTIFY use of data with less than 90% data recovery (e.g., data available for each month of the year). SUBSTITUTE data to complete the data set using interpolation techniques or techniques from regional recognized sources (e.g., government weather service stations) where on-site meteorological data are not available. See Note RC-N-8	COMPILE hourly meteorological data including rainfall that has a combined data recovery at or above 90% for the period of record. For missing data, USE data from a different tower elevation or co-located tower (if available), adjusted to complete the database. SUBSTITUTE data to complete the data set using interpolation techniques, substitution techniques, or techniques from regional recognized sources (e.g., government weather service stations) where on-site meteorological data are not available. ENSURE that the substitution process to make such determinations in accordance with Requirement RCME-A8 has been reviewed by a qualified meteorologist or professional with equivalent training or experience. INCLUDE in this review consideration of the terrain, presence of nearby water bodies, and other meteorological phenomena that may affect airflow trajectories. See Note RC-N-8
RCME-A4	JUSTIFY that the accuracy of compiled meteorological data is sufficient for the desired application. See Note RC-N-8	COMPILE meteorological data that has been collected under a qualified system of calibrations, maintenance activities, and instrument exposure. See Note RC-N-8

Table 4.3.17.3-2 Supporting Requirements for HLR-RCME-A (Cont'd)

Accurate Meteorological Data from spatially representative location(s) shall be compiled. (HLR-RCME-A)

Index No. RCME-A	Capability Category I	Capability Category II
RCME-A5	EXTRACT the following sequential hourly meteorological parameter measurements: *(a)* wind speed and direction at approximately the 10-meter level; and *(b)* some measurement or observation that can be used to determine the atmospheric stability class (see RCME-A7).	EXTRACT the following sequential hourly meteorological parameter measurements: *(a)* wind speed and direction at approximately the 10-meter level; *(b)* some measurement or observation that can be used to determine the atmospheric stability class (see RCME-A7); and *(c)* precipitation.
RCME-A6	COMPILE seasonal regional afternoon mixing height from regional data from recognized sources. See Note RC-N-9	COMPILE seasonal morning and afternoon mixing heights determined from regional data from recognized sources. See Note RC-N-9
RCME-A7	USE a simplified stability classification approach. See Note RC-N-10	USE a stability classification method from recognized sources. See Note RC-N-11
RCME-A8	REVIEW meteorological data for its accuracy to determine adequacy of data recovery and its validity.	
RCME-A9	IDENTIFY and CHARACTERIZE parameter uncertainties associated with the meteorological data in a manner that supports the applicable requirements of RCQ-C2.	
RCME-A10	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives for meteorological data in a manner that supports the applicable requirements of HLR-RCQ-C.	
RCME-A11	For PRAs on plants using a bounding site, IDENTIFY assumptions made due to the lack of site details related to meteorological data. See Note RC-N-5	

Table 4.3.17.3-3 Supporting Requirements for HLR-RCME-B

The documentation of the Meteorological Data shall provide traceability of the work. (HLR-RCME-B)

Index No. RCME-B	Capability Category I	Capability Category II
RCME-B1	DOCUMENT the process used in the Meteorological Data, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* source of data (including reasons for selection); *(b)* quality assessment; *(c)* levels of sensors; *(d)* exposure of tower; *(e)* calibration records; *(f)* period of record; *(g)* percent data recovery; and *(h)* parameter uncertainty characterization.	
RCME-B2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirement RCME-A10) associated with developing the Meteorological Data.	
RCME-B3	For PRAs conducted on a bounding site, DOCUMENT assumptions made due to the lack of site details associated with developing the Meteorological Data. See RCME-A11 See Note RC-N-5	

4.3.17.4 Objectives and Technical Requirements for Atmospheric Transport and Dispersion (RCAD)

The objectives of the Atmospheric Transport and Dispersion subelement ensure that

(a) atmospheric transport and dispersion conditions are modeled for the site;
(b) appropriate databases are utilized for a probabilistic analysis;
(c) event specific input parameters are provided;
(d) temporal and spatial changes in meteorological data are accommodated;
(e) deposition of radionuclides is included; and
(f) the Atmospheric Transport and Dispersion analysis is documented to provide traceability of the work.

Table 4.3.17.4-1 High Level Requirements for Atmospheric Transport and Dispersion

Designator	Requirement
HLR-RCAD-A	The analysis shall model the atmospheric transport and dispersion conditions at the site.
HLR-RCAD-B	The analysis shall include use of meteorological data to provide probabilistic results.
HLR-RCAD-C	The analysis shall model atmospheric transport and dispersion for input parameters and event sequence-specific input parameters specific to the site that are provided as output from the Mechanistic Source Term Analysis.
HLR-RCAD-D	The analysis shall accommodate temporal changes in meteorological conditions.
HLR-RCAD-E	The analysis shall include calculation of deposition of radionuclide particles.
HLR-RCAD-F	The documentation of the Atmospheric Transport and Dispersion modeling shall provide traceability of the work.

Table 4.3.17.4-2 Supporting Requirements for HLR-RCAD-A

The analysis shall model the atmospheric transport and dispersion conditions at the site. (HLR-RCAD-A)

Index No. RCAD-A	Capability Category I	Capability Category II
RCAD-A1	USE a straight-line steady-state Gaussian transport and dispersion model.	USE a Gaussian transport and dispersion model or similar model with temporal variations in the meteorological data that accounts for off-centerline concentrations (e.g., segmented plume model).
RCAD-A2	CALCULATE atmospheric transport and dispersion using a steady-state model (i.e., no time dependency).	CALCULATE atmospheric transport and dispersion with updates of wind speed, stability, and precipitation on a one-hour time scale.
RCAD-A3	USE a model that calculates centerline concentration and deposition. SPECIFY the spatial dimensions.	USE a model that calculates concentration and deposition on a two-dimensional grid in reasonably fine geographical areas around the site. JUSTIFY the spatial grid dimensions (e.g., includes distance for results of interest, validity of the model at outer distance).
RCAD-A4	USE a model that includes uniform hourly wind field data from a single representative meteorological tower.	
RCAD-A5	USE a model that includes wind measurements that are reasonably representative of plume travel speed and/or release height.	
RCAD-A6	IDENTIFY and CHARACTERIZE parameter uncertainties associated with the atmospheric transport and dispersion conditions in a manner that supports the applicable requirements of RCQ-C2.	
RCAD-A7	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives for atmospheric transport and dispersion conditions in a manner that supports the applicable requirements HLR-RCQ-C.	
RCAD-A8	For PRAs performed on a bounding site, IDENTIFY assumptions made due to the lack of site details that influence the atmospheric transport and dispersion conditions. See Note RC-N-5	

Table 4.3.17.4-3 Supporting Requirements for HLR-RCAD-B

The analysis shall include use of meteorological data to provide probabilistic results. (HLR-RCAD-B)

Index No. RCAD-B	Capability Category I	Capability Category II
RCAD-B1	USE meteorological data developed per the SRs of HLR-RCME-A.	
RCAD-B2	DETERMINE bounding meteorological conditions to be used in the analysis (e.g., 5th percentile dispersion factor).	USE a sampling technique (e.g., Monte Carlo method, Latin Hypercube Sampling). JUSTIFY that the sampling technique does not appreciably alter the results of interest (e.g., demonstrate the mean results vary by less than 10% compared with mean values if all meteorological data are used).

Table 4.3.17.4-4 Supporting Requirements for HLR-RCAD-C

The analysis shall model atmospheric transport and dispersion for input parameters and event sequence-specific input parameters specific to the site that are provided as output from the Mechanistic Source Term Analysis. (HLR-RCAD-C)

Index No. RCAD-C	Capability Category I	Capability Category II
RCAD-C1	USE dispersion algorithms that characterize atmospheric transport and dispersion from elevated release heights, such as the tops of buildings or stacks.	
RCAD-C2	DO NOT TAKE CREDIT for plume rise.	USE plume rise algorithms that compute the increase in elevation of the plume above its release point due to momentum (i.e., exit velocity from a vent) and/or thermal buoyancy effects (i.e., heated discharges). See Note RC-N-12
RCAD-C3	USE algorithms that account for building wake effects. See Note RC-N-13	
RCAD-C4	IDENTIFY and CHARACTERIZE parameter uncertainties associated with the modeling of atmospheric transport and dispersion in a manner that supports the applicable requirements of RCQ-C2.	
RCAD-C5	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives for modeling atmospheric transport and dispersion input parameters in a manner that supports the applicable requirements HLR-RCQ-C.	
RCAD-C6	For PRAs performed on a bounding site, IDENTIFY assumptions made due to the lack of site details that influence the input parameters for atmospheric transport and dispersion. See Note RC-N-5	

Table 4.3.17.4-5 Supporting Requirements for HLR-RCAD-D

The analysis shall accommodate temporal changes in meteorological conditions. (HLR-RCAD-D)

Index No. RCAD-D	Capability Category I	Capability Category II
RCAD-D1	USE a transport and dispersion model without spatial or temporal meteorological variability.	USE a transport and dispersion model that incorporates varying meteorology and straight-line direction for each release time period (i.e., segmented plume).
RCAD-D2	USE a transport and dispersion model with a single plume.	USE a transport and dispersion model with multiple plumes.
RCAD-D3	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives for temporal and spatial changes in meteorological conditions in a manner that supports the applicable requirements of HLR-RCQ-C.	
RCAD-D4	For PRAs performed on a bounding site, IDENTIFY assumptions made due to the lack of site details that influence the temporal and spatial changes in meteorological conditions. See Note RC-N-5	

Table 4.3.17.4-6 Supporting Requirements for HLR-RCAD-E

The analysis shall include calculation of deposition of radionuclide particles. (HLR-RCAD-E)

Index No. RCAD-E	Capability Category I	Capability Category II
RCAD-E1	INCLUDE a single dry-deposition velocity for aerosols in the model.	INCLUDE a dry-deposition velocity for each aerosol particle size in the model.
RCAD-E2	DO NOT INCLUDE wet deposition.	INCLUDE wet deposition of aerosol particles for various precipitation intensities in the model. See Note RC-N-14
RCAD-E3	DO NOT INCLUDE wet or dry deposition source depletion in the model.	INCLUDE in the model either *(a)* dry deposition and JUSTIFY the exclusion of wet deposition; or *(b)* both dry and wet deposition.
RCAD-E4	DO NOT INCLUDE resuspension of deposited radionuclide particles in the model.	INCLUDE resuspension of deposited radionuclide particles in the model. See Note RC-N-15
RCAD-E5	IDENTIFY and CHARACTERIZE parameter uncertainties associated with the deposition of radionuclide particles in a manner that supports the applicable requirements of RCQ-C2.	
RCAD-E6	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives for deposition of radionuclide particles in a manner that supports the applicable requirements HLR-RCQ-C.	
RCAD-E7	For PRAs performed for a bounding site, IDENTIFY assumptions made due to the lack of site details that influence the deposition of radionuclide particles. See Note RC-N-5	

Table 4.3.17.4-7 Supporting Requirements for HLR-RCAD-F

The documentation of the Atmospheric Transport and Dispersion modeling shall provide traceability of the work. (HLR-RCAD-F)

Index No. RCAD-F	Capability Category I	Capability Category II
RCAD-F1	DOCUMENT the process used in the Atmospheric Transport and Dispersion modeling, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* atmospheric transport and dispersion model; *(b)* calculation grid; *(c)* time scale; *(d)* meteorological sampling method; *(e)* plant/site characteristics (e.g., release height, building dimensions); *(f)* parameter uncertainty characterization.	
RCAD-F2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirements RCAD-A7, RCAD-C5, RCAD-D3, and RCAD-E6) associated with the Atmospheric Transport and Dispersion modeling.	
RCAD-F3	For PRAs performed on a bounding site, DOCUMENT assumptions and limitations due to the lack of site details associated with the Atmospheric Transport and Dispersion modeling. See RCAD-A8, RCAD-C6, RCAD-D4, and RCAD-E7 See Note RC-N-5	

4.3.17.5 Objectives and Technical Requirements for Dosimetry (RCDO)

The objectives of the Dosimetry subelement ensure that

(a) applicable exposure pathways and the effects of mitigation actions are included;
(b) dose conversion factors are applied from recognized sources;
(c) the Dosimetry analysis is documented to provide traceability of the work.

Table 4.3.17.5-1 High Level Requirements for Dosimetry

Designator	Requirement
HLR-RCDO-A	The analysis shall include applicable exposure pathways including cloudshine, groundshine, skin deposition, inhalation and ingestion, and the effect of mitigation actions on received dose.
HLR-RCDO-B	The analysis shall apply dose conversion factors (DCFs) from recognized sources.
HLR-RCDO-C	The documentation of the Dosimetry modeling shall provide traceability of the work.

Table 4.3.17.5-2 Supporting Requirements for HLR-RCDO-A

The analysis shall include applicable exposure pathways including cloudshine, groundshine, skin deposition, inhalation and ingestion, and the effect of mitigation actions on received dose. (HLR-RCDO-A)

Index No. RCDO-A	Capability Category I	Capability Category II
RCDO-A1	IDENTIFY the exposure pathways used in the analysis. JUSTIFY excluding any of the following pathways: *(a)* cloudshine; *(b)* groundshine; *(c)* skin deposition; *(d)* inhalation; *(e)* ingestion. See Note RC-N-16	
RCDO-A2	USE the plume concentrations and deposition resulting from the atmospheric, transport, and dispersion model to calculate doses over the exposure period(s) specified in RCDO-A3.	
RCDO-A3	SPECIFY and JUSTIFY the exposure period(s) used in the analysis (e.g., exposure periods are consistent with objectives of the analysis).	
RCDO-A4	USE a semi-infinite cloud immersion dose model to determine dose.	USE a semi-infinite plume model with correction factor or a finite plume model to account for the dimensions of the plume in determining the deep dose.
RCDO-A5	USE a model that integrates groundshine exposure over the appropriate time period (e.g., accounting for deposited materials both during and after plume passage).	
RCDO-A6	DO NOT INCLUDE skin deposition pathway in the model.	INCLUDE skin deposition and beta exposure to the skin from the plume in the model.
RCDO-A7	USE a generic breathing rate for the population.	USE and JUSTIFY breathing rates for each specified cohort (e.g., breathing rates for the anticipated activities of the cohort).
RCDO-A8	DO NOT INCLUDE ingestion pathways in the model.	USE generic intake quantities of foodstuffs and water.
RCDO-A9	IDENTIFY and CHARACTERIZE parameter uncertainties associated with the exposure pathways in a manner that supports the applicable requirements of RCQ-C2.	
RCDO-A10	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives for exposure pathways in a manner that supports the applicable requirements of HLR-RCQ-C.	

Table 4.3.17.5-3 Supporting Requirements for HLR-RCDO-B

The analysis shall apply DCFs from recognized sources. (HLR-RCDO-B)

Index No. RCDO-B	Capability Category I	Capability Category II
RCDO-B1	USE effective DCFs from recognized sources. See Note RC-N-17	USE organ-specific DCFs from recognized sources. See Note RC-N-17
RCDO-B2	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives for DCFs in a manner that supports the applicable requirements of HLR-RCQ-C. See Note RC-N-18	

Table 4.3.17.5-4 Supporting Requirements for HLR-RCDO-C

The documentation of the Dosimetry modeling shall provide traceability of the work. (HLR-RCDO-C)

Index No. RCDO-C	Capability Category I	Capability Category II
RCDO-C1	DOCUMENT the process used in the dose analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* exposure pathways models; *(b)* recognized sources used for DCFs; *(c)* protection factors; *(d)* parameter uncertainty characterization.	
RCDO-C2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in RCDO-A10 and RCDO-B2) associated with the dose analysis.	

4.3.17.6 Objectives and Technical Requirements for Health Effects (RCHE)

The objectives of the Health Effects subelement ensure that

(a) modeling the health effects is conducted for the chosen parameters as a function of dose and dose rate;

(b) sources of health effects from exposure to ionizing radiation are based on recommendations from national or international bodies;

(c) the Health Effects modeling is documented to provide traceability of the work.

Table 4.3.17.6-1 High Level Requirements for Health Effects

Designator	Requirement
HLR-RCHE-A	Each health effect parameter that is chosen shall be clearly defined in terms of the models of the risk of health effects as a function of dose and dose rate.
HLR-RCHE-B	The risk models of health effects versus dose and dose rate shall be based on recommendations of the international or national bodies or national regulatory agencies.
HLR-RCHE-C	The documentation of the Health Effects modeling shall provide traceability of the work.

Table 4.3.17.6-2 Supporting Requirements for HLR-RCHE-A

Each health effect parameter that is chosen shall be clearly defined in terms of the models of the risk of health effects as a function of dose and dose rate. (HLR-RCHE-A)

Index No. RCHE-A	Capability Category I	Capability Category II
RCHE-A1	IDENTIFY the early and latent health effects for which parameters are needed. See Note RC-N-19	
RCHE-A2	INCLUDE early health effect input parameters based on a simplified set of organs and/or a reduced set of radionuclides (e.g., I-131 equivalent).	INCLUDE the early health effect input parameters (e.g., dose-response parameters for a hazard function) required for the target organ of the body involved.
RCHE-A3	INCLUDE latent health effect input parameters based on a simplified set of organs [e.g., total effective dose equivalent (TEDE)] and/or a reduced set of radionuclides (e.g., I-131 equivalent).	INCLUDE the latent health effect input parameters (e.g., dose and dose-rate effectiveness factors, cancer-incidence risk factors, and cancer-fatality risk factors) required for the target organ of the body involved.
RCHE-A4	USE homogenous health effect input parameters related to age and gender attributes.	
RCHE-A5	IDENTIFY and CHARACTERIZE parameter uncertainties associated with the health effect parameters in a manner that supports the applicable requirements of RCQ-C2.	
RCHE-A6	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives for health effect parameters in a manner that supports the applicable requirements of HLR-RCQ-C.	
RCHE-A7	For PRAs performed using a bounding site, IDENTIFY assumptions made due to the lack of site details that influence the health effect parameters. See Note RC-N-5	

Table 4.3.17.6-3 Supporting Requirements for HLR-RCHE-B

The risk models of health effects versus dose and dose rate shall be based on recommendations of the international or national bodies or national regulatory agencies. (HLR-RCHE-B)

Index No. RCHE-B	Capability Category I	Capability Category II
RCHE-B1	USE risk factors recommended by internationally recognized agencies to model the risk factor input parameters. See Note RC-N-20	
RCHE-B2	IDENTIFY and CHARACTERIZE parameter uncertainties associated with the risk model of health effects in a manner that supports the applicable requirements of RCQ-C2.	
RCHE-B3	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives for the risk model of health effects in a manner that supports the applicable requirements of HLR-RCQ-C.	

Table 4.3.17.6-4 Supporting Requirements for HLR-RCHE-C

The documentation of the Health Effects modeling shall provide traceability of the work. (HLR-RCHE-C)

Index No. RCHE-C	Capability Category I	Capability Category II
RCHE-C1	DOCUMENT the process used in the Health Effects analysis, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* descriptions of target organs selected for early/latent fatality/injury models; *(b)* sources utilized for health risk models (e.g., BEIR, International Commission on Radiological Protection) and the version used (e.g., BEIR VII [RC-10]); *(c)* parameter uncertainty characterization.	
RCHE-C2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in RCHE-A6 and RCHE-B3) associated with the risk models of health effects.	
RCHE-C3	For PRAs conducted on a bounding site, DOCUMENT assumptions and limitations due to the lack of site details associated with the risk models of health effects. See RCHE-A7 See Note RC-N-5	

4.3.17.7 Objectives and Technical Requirements for Economic Factors (RCEC)

The objectives of the Economic Factors subelement ensure that

(a) the economic parameters are defined;
(b) appropriate data for cost parameter estimates are used; and
(c) the Economic Factors analysis is documented to provide traceability of the work.

Table 4.3.17.7-1 High Level Requirements for Economic Factors

Designator	Requirement
HLR-RCEC-A	Each economic parameter shall be clearly defined in terms of the model.
HLR-RCEC-B	Parameter estimates shall be based on relevant generic data or data specific to the site and regional data consistent with the parameter definitions of the SRs of HLR-RCEC-A.
HLR-RCEC-C	The documentation of the Economic Factors modeling shall provide traceability of the work.

Table 4.3.17.7-2 Supporting Requirements for HLR-RCEC-A

Each economic parameter shall be clearly defined in terms of the model. (HLR-RCEC-A)

Index No. RCEC-A	Capability Category I	Capability Category II
RCEC-A1	IDENTIFY the cost categories for which parameter estimates are required. Examples of cost categories include the following: *(a)* evacuation costs; *(b)* relocation costs, including temporary unemployment; *(c)* land value; *(d)* depreciation; *(e)* crop losses; *(f)* decontamination costs; *(g)* loss of use of off-site property; and *(h)* medical costs (e.g., costs estimated based on population dose). See Note RC-N-21	
RCEC-A2	IDENTIFY economic model parameters for the identified cost categories of Requirement RCEC-A1.	

Table 4.3.17.7-3 Supporting Requirements for HLR-RCEC-B

Parameter estimates shall be based on relevant generic data or data specific to the site and regional data consistent with the parameter definitions of the SRs of HLR-RCEC-A. (HLR-RCEC-B)

Index No. RCEC-B	Capability Category I	Capability Category II
RCEC-B1	ENSURE that the economic modeling parameter estimates are consistent with the parameter definitions established in Requirements RCEC-A1 and RCEC-A2.	
RCEC-B2	ESTIMATE cost parameter values using regional data applicable to the site and generic data (as needed).	
RCEC-B3	USE recognized sources of cost data or JUSTIFY the use of generic data. See Note RC-N-22	
RCEC-B4	ENSURE cost parameter values incorporate the time frame of interest (e.g., consumer price index adjustment to account for inflation). See Note RC-N-22	
RCEC-B5	IDENTIFY and CHARACTERIZE parameter uncertainties associated with the economic factors in a manner that supports the applicable requirements of RCQ-C2.	
RCEC-B6	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives for economic factors in a manner that supports the applicable requirements of HLR-RCQ-C.	
RCEC-B7	For PRAs prior to site selection, IDENTIFY assumptions made due to the lack of site details that influence the economic factors. See Note RC-N-5	

Table 4.3.17.7-4 Supporting Requirements for HLR-RCEC-C

The documentation of the Economic Factors modeling shall provide traceability of the work. (HLR-RCEC-C)

Index No. RCEC-C	Capability Category I	Capability Category II
RCEC-C1	DOCUMENT the process used in the Economic Factors, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* parameter definitions; *(b)* generic sources used; *(c)* sources used specific to the site; *(d)* time period of sources (e.g., most recent census); *(e)* adjustments to parameter estimates [e.g., consumer price index (CPI) adjustment]; and *(f)* parameter uncertainty characterization.	
RCEC-C2	DOCUMENT the Economic Factors sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in RCEC-B6) associated with the Economic Factors analysis. See RCEC-B6	
RCEC-C3	For PRAs performed on a bounding site, DOCUMENT assumptions and limitations due to the lack of site details associated with the Economic Factors analysis. See Note RC-N-5	

4.3.17.8 Objectives and Technical Requirements for Consequence Quantification (RCQ)

The objectives of the Consequence Quantification subelement ensure that

(a) appropriate models and computer codes are used to perform a radiological consequence analysis;
(b) reviews of the Radiological Consequence Analysis results are used to identify risk-significant contributors;
(c) uncertainties and their impacts are characterized; and
(d) the Consequence Quantification analysis is documented to provide traceability of the work.

Table 4.3.17.8-1 High Level Requirements for Consequence Quantification

Designator	Requirement
HLR-RCQ-A	Quantification shall use appropriate models and codes and shall account for method-specific limitations and features.
HLR-RCQ-B	Quantification results shall be reviewed, and risk-significant contributors shall be identified. The results should be traceable to the inputs and assumptions.
HLR-RCQ-C	Uncertainties in the results shall be characterized, and the potential impact on the results shall be reported.
HLR-RCQ-D	The documentation of the Consequence Quantification results (output) shall provide traceability of the work.

Table 4.3.17.8-2 Supporting Requirements for HLR-RCQ-A

Quantification shall use appropriate models and codes and shall account for method-specific limitations and features. (HLR-RCQ-A)

Index No. RCQ-A	Capability Category I	Capability Category II
RCQ-A1	For each release category, PERFORM quantification using computer codes that have been demonstrated to generate appropriate results when compared to accepted algorithms (e.g., Gaussian plume model).	
RCQ-A2	IDENTIFY features and limitations of models and codes that could impact the results. Examples include the following: *(a)* temporal regime-minimum/maximum plume durations; *(b)* spatial regime-minimum/maximum distances, flat earth versus terrain impacts; *(c)* parameter limits. JUSTIFY method-specific features and limitations, as needed, that could impact results.	
RCQ-A3	COMPILE list of event sequence families and associated radiological consequences.	

Table 4.3.17.8-3 Supporting Requirements for HLR-RCQ-B

Quantification results shall be reviewed, and risk-significant contributors shall be identified. The results should be traceable to the inputs and assumptions. (HLR-RCQ-B)

Index No. RCQ-B	Capability Category I	Capability Category II
RCQ-B1	REVIEW output files for indications of improper quantification (e.g., error statements, warning statements, and unexpected results, such as zero values). JUSTIFY acceptance of any indications of code execution errors (e.g., document evaluation of error messages and why results are not materially impacted).	
RCQ-B2	CONFIRM appropriate modeling and code execution by examining the results. See Note RC-N-23	
RCQ-B3	IDENTIFY risk-significant contributors (elements) using the SRs of HLR-RI-B. See Note RC-N-24	

Table 4.3.17.8-4 Supporting Requirements for HLR-RCQ-C

Uncertainties in the results shall be characterized, and the potential impact on the results shall be reported. (HLR-RCQ-C)

Index No. RCQ-C	Capability Category I	Capability Category II
RCQ-C1	ASSESS the consequence effects of the model uncertainties, related assumptions, and reasonable alternatives, identified for each technical subelement in Radiological Consequence Analysis by performing a qualitative or quantitative evaluation of the effects of the individual sources of uncertainty or of combinations of interest on each modeled consequence metric.	
RCQ-C2	CHARACTERIZE the uncertainty distribution of each modeled consequence metric for each modeled event sequence family consistent with the characterization of parameter uncertainties. See Requirements RCRE-A2, RCPA-A12, RCPA-B6, RCME-A9, RCAD-A6, RCAD-C4, RCDO-A9, RCHE-A5, and RCEC-B5, as applicable	CALCULATE the uncertainty distribution of each modeled consequence metric for each modeled event sequence family by propagating the uncertainty distributions on the parameters for the risk-significant contributors, and those model uncertainties explicitly characterized by a probability distribution in such as a way that the dependencies between uncertain phenomena are accounted for. See Requirements RCRE-A2, RCPA-A12, RCPA-B6, RCME-A9, RCAD-A6, RCAD-C4, RCDO-A9, RCHE-A5, and RCEC-B5, as applicable

Table 4.3.17.8-5 Supporting Requirements for HLR-RCQ-D

The documentation of the Consequence Quantification results (output) shall provide traceability of the work. (HLR-RCQ-D)

Index No. RCQ-D	Capability Category I	Capability Category II
RCQ-D1	DOCUMENT the process used in the Consequence Quantification, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* computer codes used; *(b)* general description of quantification process; *(c)* assumptions; *(d)* results (e.g., early health effects, latent health effects, economic factors); *(e)* description of results review; *(f)* results of pertinent sensitivity cases; *(g)* uncertainty discussion.	
RCQ-D2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in Requirements RCQ-A2, RCQ-C1, and RCQ-C2) associated with the Consequence Quantification.	
RCQ-D3	DOCUMENT limitations in the quantification process that would impact applications.	

4.3.17.9 Peer Review Requirements for Radiological Consequence Analysis

4.3.17.9.1 Purpose

This Section provides requirements for peer review of the Radiological Consequence Analysis element of the PRA.

4.3.17.9.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the area of radiological consequence analysis. The Radiological Consequence Analysis element review shall coordinate, whether through shared team members or other approaches, with the review of the Mechanistic Source Term Analysis and Risk Integration elements to ensure consistency between the release categories, the modeling approaches, and desired outputs.

4.3.17.9.3 Review of Radiological Consequence Analysis to Confirm the Methodology

A review shall be performed of the calculated Radiological Consequence Analysis, including any subelements within the PRA scope.

For the Release Category to Radiological Consequence subelement, the review should include the following:

(a) the consistency and traceability between the Event Sequence Analysis, the release categories defined in the Mechanistic Source Term Analysis, and the Radiological Consequence Analysis;

(b) proper selection of release characteristics to be modeled for the selected consequences metrics.

For the Protective Action Parameters and Other Site Data Analysis subelement, the review should include the following:

(a) use of appropriate protective action characteristics and population/land use/geography information.

For the Meteorological Data subelement, the review should include the following:

(a) use of appropriate meteorological data.

For the Atmospheric Transport and Dispersion subelement, the review should include the following:

(a) use of appropriate atmospheric dispersion model, including consideration of the selected consequence metric and nature of release and plant site;

(b) use of appropriate release and plume characteristics, including consideration of the plant design, release scenario, and selected location;

(c) use of appropriate atmospheric dispersion modeling parameters.

For the Dosimetry subelement, the review should include the following:

(a) the consideration of all applicable exposure pathways and their associated modeling parameters;

(b) use of appropriate dose DCFs.

For the Health Effects subelement, the review should include the following:

(a) the selection of modeled health effects, in consideration of the selected consequence metric;

(b) use of appropriate health effect factors.

For the Economic Factors subelement, the review should include the following:

(a) proper identification of economic factors to be included in the analysis;

(b) use of appropriate data and models.

For the Consequence Quantification subelement, the review should include the following:

(a) use of appropriate uncertainty quantification methods and models;

(b) the review and identification of risk-significant contributors;

(c) the treatment of model uncertainty.

4.3.17.10 References for Radiological Consequence Analysis

The following is a list of publications referenced in this Standard.

[RC-1] ASME/ANS RA-S-1.3-2017, "Standard for Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications," American Society of Mechanical Engineers/American Nuclear Society, 2013

[RC-2] NUREG/CR-2300, J. HICKMAN et al., "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," American Nuclear Society/Institute of Electrical and Electronic Engineers/U.S. Nuclear Regulatory Commission, 1983

[RC-3] NUREG/CR-6863, SAND2004-5900, "Development of Evacuation Time Estimate Studies for Nuclear Power Plants," Sandia National Laboratories/U.S. Nuclear Regulatory Commission, January 2005

[RC-4] Advanced Light Water Reactor Utility Requirements Document, Vol. 1, "ALWR Policy and Summary of Top-Tier Requirements," Rev. 2, Electric Power Research Institute, March 1999

[RC-5] NUREG/CR-2239, SAND81-1549, "Technical Guidance for Siting Criteria Development," Sandia National Laboratories, U.S. Nuclear Regulatory Commission, December 1982

[RC-6] ANSI/ANS-3.11-2015 (R2020), "Determining Meteorological Information at Nuclear Facilities," American Nuclear Society, La Grange Park, Illinois, 2015

[RC-7] G. C. HOLZWORTH, "Mixing Heights, Wind Speeds, and Potential for Urban Air Pollution Throughout the Contiguous United States," Office of Air Programs Publication AP-101, U.S. Environmental Protection Agency, 1972

[RC-8] Regulatory Guide 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Rev. 1, U.S. Nuclear Regulatory Commission, March 2007

[RC-9] NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," U.S. Nuclear Regulatory Commission, July 1982

[RC-10] "BEIR VII: Health Risks from Exposure to Low Levels of Ionizing Radiation," National Research Council, National Academies Press, Washington, D.C., 2006

[RC-11] ICRP Publication 103: "The 2007 Recommendations of the International Commission on Radiological Protection," Vol. 37, No. 2-4, International Commission on Radiological Protection, 2007

[RC-12] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," U.S. Nuclear Regulatory Commission, December 1990

[RC-13] ICRP 60, "1990 Recommendations of the International Commission on Radiological Protection," Annuals of the ICRP, Ann. ICRP 21 (1-3), International Commission on Radiological Protection, 1991

[RC-14] ICRP 72, "Age-dependent Doses to the Members of the Public from Intake of Radionuclides: Part 5, Compilation of Ingestion and Inhalation Coefficients," Annuals of the ICRP, Ann. ICRP 26 (1), P 072, errata in SG 03 JAICRP 32(1-2), International Commission on Radiological Protection, 1995

[RC-15] FGR-11, K. F. Eckerman, A. B. Wolbarst, and C. B. Richardson, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," U.S. Environmental Protection Agency/Oak Ridge National Laboratory, September 1988

[RC-16] FGR-12, K. F. Eckerman and J. C. Ryman, "External Exposure to Radionuclides in Air, Water, and Soil," U.S. Environmental Protection Agency, September 1993

[RC-17] FGR-13, "Cancer Risk Coefficients for Environmental Exposure to Radionuclides: Updates and Supplements," U.S. Environmental Protection Agency, 2006

[RC-18] AMS 1977, "American Meteorological Society Workshop on Stability Classification Schemes and Sigma Curves – Summary and Recommendations," Bulletin of the American Meteorological Society, Vol. 58, 1977

[RC-19] D. B. Turner, "Workbook of Atmospheric Dispersion Estimates," PSH-999-AP-26, U.S. Environmental Protection Agency, 1970

[RC-20] Regulatory Guide 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Rev. 1, U.S. Nuclear Regulatory Commission, March 2007

[RC-21] EPA-454/R-99-005, "Meteorological Monitoring Guidance for Regulatory Modeling Applications," U.S. Environmental Protection Agency, February 2000

[RC-22] D. H. Slade, ed., "Meteorology and Atomic Energy 1968," TID-24190, U.S. Atomic Energy Commission, July 1968

[RC-23] D. Randerson, "Atmospheric Transport and Diffusion Consequence Assessment Models," FOE-TIC-27601, U.S. Department of Energy, July 1984

[RC-24] Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," U.S. Nuclear Regulatory Commission, November 1982, correction issued February 1983

[RC-25] Regulatory Guide 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments," U.S. Nuclear Regulatory Commission, June 2003

[RC-26] G. A. Briggs, "Plume Rise Predictions. In: Lectures on Air Pollution and Environmental Impact Analyses," American Meteorological Society, Boston, MA

[RC-27] W. G. N. Slinn, "Some Approximations for the Wet and Dry Removal of Particles and Gases from the Atmosphere," Water, Air, & Soil Pollution, Vol. 7, Issue 4, pp. 513–543, 1977

[RC-28] W. G. N. Slinn, "Parameterizations for Resuspension and for Wet and Dry Deposition of Particles and Gases for use in Radiation Dose Calculations," Nuclear Safety, Vol. 19, pp. 205–219, 1978

[RC-29] G. A. Loosemore, "Evaluation and Development of Models for Resuspension of Aerosols at Short Times after Deposition," UCRL-JC-149850, Lawrence Livermore National Laboratory, August 2002

[RC-30] L. R. Anspaugh, et al., "Resuspension and Redistribution of Plutonium in Soils," Health Physics, pp. 571–582, 1975

[RC-31] NUREG/CR-7110, "State-of-the-Art Reactor Consequence Analyses Project," U.S. Nuclear Regulatory Commission, May 2013

NONMANDATORY APPENDIX RC: NOTES AND EXPLANATORY MATERIAL FOR RADIOLOGICAL CONSEQUENCE ANALYSIS

RC.1 NOTES ASSOCIATED WITH RADIOLOGICAL CONSEQUENCE ANALYSIS

Table RC-1 Notes Supporting Radiological Consequence Analysis Requirements

Number	Notes
RC-N-1	The description of a bounding site can be the identification of an existing site if that site bounds the radiological consequences for other sites under consideration. See RCRE-A1
RC-N-2	See NUREG-1150 [RC-12] and NUREG/CR-7110 [RC-31] for examples of generic data sources. See RCPA-A5, RCPA-A7
RC-N-3	This Supporting Requirement (SR) is not applicable to operating plants. See RCPA-A14
RC-N-4	"Local" refers to the geographical area associated with the plume exposure pathway emergency planning zone (e.g., approximately 10-mile radius). "Regional" refers to the geographical area evaluated in the model that is beyond the local area (e.g., 10- to 50-mile radius). See RCPA-B1
RC-N-5	This SR is not applicable to PRAs performed for a specific site. See RCPA-B5, RCPA-B8, RCPA-C3, RCME-A11, RCME-B3, RCAD-A8, RCAD-C6, RCAD-D4, RCAD-E7, RCAD-F3, RCHE-A7, RCHE-C3, RCEC-B7, RCEC-C3
RC-N-6	Data from airports may be inadequate for radiological consequence assessment. The reported wind speeds may only be accurate down to one mph. Many airport records do not have adequate procedures for reporting lower speeds or may not have anemometers that are sensitive at low wind speeds. In addition, there is often a runway direction bias in older manually recorded wind direction observations. Lastly, the technique for typing atmospheric turbulence into stability classes results in a larger frequency of slightly stable and neutral stability and a lower frequency of very unstable and very stable conditions. See RCME-A1
RC-N-7	Factors to be assessed may include proximity to the site, exposure of the site to local influences (i.e., terrain-induced effects, such as river-valley orientation; nearness to large bodies of water), long-term climatology (e.g., wind direction frequencies, wind speed averages, and stability category averages (e.g., AMS 1977 [RC-18]), and poor data recovery rate. See RCME-A1
RC-N-8	ANSI/ANS-3.11-2015 (R2020) [RC-6] provides information on qualified meteorologists and data substitution. Additionally, Table 1 of ANSI/ANS-3.11-2015 (R2020) establishes accuracies for each parameter. See RCME-A3, RCME-A4
RC-N-9	An example of a recognized source of seasonal data is Holzworth 1972 [RC-7]. See RCME-A6
RC-N-10	An example of a simplified approach is the stability array (STAR) method (Turner 1970 [RC-19]). See RCME-A7
RC-N-11	Examples of recognized sources include the following: *(a)* delta-T and the table for converting to stability class (RG 1.23 [RC-20]); and *(b)* sigma-theta and the table for converting to stability class (ANSI/ANS-3.11-2015 (R2020) [RC-6]) using the U.S. Environmental Protection Agency (EPA) correction (EPA-454 [RC-21]) for nighttime hours. See RCME-A7

Table RC-1 Notes Supporting Radiological Consequence Analysis Requirements (Cont'd)

Number	Notes
RC-N-12	Acceptable plume rise algorithms can be found in Briggs 1975 [RC-26]. See RCAD-C2
RC-N-13	Acceptable algorithms that account for building wake effects are found in Slade 1968 [RC-22], Randerson 1984 [RC-23], RG 1.145 [RC-24], and RG 1.194 [RC-25]. See RCAD-C3
RC-N-14	Examples of wet deposition can be found in Slinn 1977 [RC-27] and Randerson 1984 [RC-23]. See RCAD-E2
RC-N-15	Examples of resuspension models can be found in Slinn 1978 [RC-28], Loosemore 2002 [RC-29], and Anspaugh 1975 [RC-30]. See RCAD-E4
RC-N-16	For example, the justification can take the form of a demonstration that the dose from an excluded pathway is small in comparison to other pathways or that the exposure pathway is incompatible with the atmospheric transport assumptions (e.g., if deposition is not modeled, then ingestion cannot be modeled.) See RCDO-A1
RC-N-17	Examples of recognized sources for DCFs include the following: *(a)* ICRP (e.g., ICRP 60 [RC-13], ICRP 72 [RC-14]), and *(b)* Federal guidance reports (FGRs) (e.g., FGR-11 [RC-15], FGR-12 [RC-16], FGR-13 [RC-17]). See RCDO-B1
RC-N-18	Modeling uncertainties regarding DCFs may include factors such as the applicability of the DCFs for the physical or chemical form of the released radionuclides and also uncertainty regarding the derivation of the DCF values. See RCDO-B2
RC-N-19	Examples of early health effects include the following: *(a)* hematopoietic syndrome (organ: bone marrow); *(b)* pulmonary syndrome (organ: lung); *(c)* gastrointestinal syndrome (organ: small intestine/colon); *(d)* prodromal syndrome (organ: abdomen); *(e)* thyroiditis/hypothyroidism (organ: thyroid); *(f)* erythema (organ: skin); *(g)* cataract (organ: lens of eye); *(h)* fetal death/microencephaly (organ: embryo). Examples of somatic latent health effects include the following: *(a)* leukemia (organ: red bone marrow); *(b)* bone cancer (organ: bone surface); *(c)* breast cancer (organ: breast); *(d)* lung cancer (organ: lung); *(e)* thyroid cancer (organ: thyroid); *(f)* gastrointestinal cancer (organ: lower large intestine); *(g)* skin cancer (organ: skin); *(h)* remainder (i.e., cancers not specifically included above). See RCHE-A1
RC-N-20	Examples of internationally recognized agency models for risk factors include the following: *(a)* BEIR V or BEIR VII [RC-10]; *(b)* ICRP 60 [RC-13] or ICRP 103 [RC-11]; *(c)* FGR-13 [RC-17]; and *(d)* UNSCEAR. See RCHE-B1

Table RC-1 Notes Supporting Radiological Consequence Analysis Requirements (Cont'd)

Number	Notes
RC-N-21	Some radiological consequence analyses may require the calculation of other economic impacts. For example, economic impacts associated with on-site losses (e.g., costs for replacement power) are not addressed in this Standard but may need to be considered. See RCEC-A1
RC-N-22	Recognized sources of cost data include those from the U.S. Department of Agriculture, U.S. Census Bureau, U.S. Department of Labor, U.S. Department of Commerce, and NUREG-1150 [RC-12]). See RCEC-B3, RCEC-B4
RC-N-23	Means of results examination may include the following: *(a)* comparing results from multiple model runs for consistency and expected trends (e.g., multiple source terms); *(b)* comparing results with results of other studies (e.g., NUREG-1150 plants [RC-12]) for reasonableness. See RCQ-B2
RC-N-24	Examples of areas that may be investigated include the following: *(a)* weather variability; *(b)* emergency response actions; *(c)* exposure pathways; *(d)* early phase versus long-term phase contributors; *(e)* population cohorts (e.g., transients); and *(f)* economic inputs (e.g., population relocation costs versus land remediation costs). See RCQ-B3

RC.2 Additional Information for Radiological Consequence Analysis

The complete range of calculations carried out in the course of a PRA bridges the gap between the engineering and operations associated with a nuclear facility and the potential risks that the facility poses to the public. Consequence analysis provides the final link in this chain of calculations and is intended to assess the effect of releases of radionuclides to the atmosphere on the surrounding population and the environment.

Consequence modeling can therefore be defined as a set of realistic calculations of the ranges (probabilities of occurrence and magnitudes) of potential adverse impacts that would follow from the dose received by humans due to a release of radionuclides. These adverse impacts, commonly referred to as “public risks,” include (a) early fatalities, (b) latent cancer fatalities, (c) early injuries, and (d) nonfatal cancers. In addition, adverse impacts can occur due to contamination of property, land, and surface water. Consequence analyses may include assessments of the economic impact of dose avoidance strategies such as relocation of population, decontamination of land and structures, and interdiction of foodstuffs.

Consequence modeling provides the means for relating these risks to the characteristics of the radioactive release and has many actual or potential applications, including the following:

(a) risk evaluation, generic or site-specific, individual or the general population;

(b) environmental impact assessment;

(c) rule-making and regulatory procedures;

(d) emergency response planning;

(e) development of criteria for the acceptability of risk;

(f) instrumentation needs and dose assessment;

(g) facility siting;

(h) comparison with safety goals evaluation;

(i) evaluation of alternative design features (e.g., Severe Accident Mitigation Alternatives);

(j) cost-benefit for backfits. A radiological consequence analysis incorporates information including demography, emergency response planning, physical properties of radionuclides, meteorology, atmospheric diffusion and transport, size of nearby structures, health physics, and other disciplines. Use of this information is detailed in this element. A prior document, NUREG/CR-2300 [RC-2], was prepared to describe a method for consequence analyses (i.e., how to do such calculations rather than what to do), which is the focus of this element.

The objective of the Radiological Consequence Analysis element requirements is to ensure that the calculation of off-site radiological consequences is performed in such a way that

(a) the scope of the consequence analysis in terms of the consequence metrics to be calculated (e.g., site boundary doses, health effects, etc.) is appropriate for the PRA life cycle-design stage and PRA applications under consideration;

(b) the transition and flow of information from the portion of the analysis that describes the event sequence, according to the requirements of Event Sequence Analysis, up to the point of determining the characteristics of the release categories, according to the requirements of Mechanistic Source Term Analysis, to the beginning of the Radiological Consequence Analysis is traceable and coherent;

(c) where applicable, the following factors are addressed:

(1) population and off-site emergency protective action parameters and other site data are properly collected and assembled;

(2) meteorological data are properly gathered and adequately represent the conditions for the site;

(3) atmospheric transport and dispersion analyses are properly conducted;

(4) dosimetry properly uses appropriate methodologies, considers all relevant pathways, and properly accounts for mitigative measures for individual and population groups;

(5) health effects are properly modeled using accepted dose-response relationships;

(6) appropriate economic factors are properly assembled and used in a consistent framework for the estimation of property damage and associated costs. Cost categories should fully envelop the possible damage types, and the origin of cost information should be fully documented.

Note that one or more of the above objectives may not be included depending on the PRA scope and Capability Category selected for the PRA applications.

The scope of the radiological consequence analysis covered by this Standard includes determination of the consequences of accidental releases of radioactive materials to the atmosphere. Limited treatment of the impact on aquatic pathways of accidental release of radioactive materials that could reach liquid pathways is included.

Radiological consequence analyses can be performed to support a spectrum of applications ranging from the determination of site boundary radiological dose, to hypothetical receptors at specific locations, to the determination of a spectrum of radiological consequences to a population and the surrounding land. The analyses may also be performed to determine whether Quantitative Health Objectives (QHOs) have been met. The analytical tools and data necessary to appropriately support these decisions may differ. It is important that the data gathered and processed, and the analytical tools used are appropriate to support the application under consideration. The scope of the radiological consequence analysis should be consistent with the scope of the overall PRA and appropriate to support the application under consideration.

A different aspect of scope has to do with whether the radiological consequence analysis will combine the entire event sequence from plant operating state and initiating event through the consequence end state. An alternative scope would have the consequence analysis focus only on that portion of the event sequence that begins with the release category and proceeds to the consequence end state. Both interpretations have historically been followed. The description of the scope must make this aspect of the analysis scope clear.

For each application, it is important that the model reasonably represent the actual as-built, as-operated nuclear installation or facility and its environs that are being analyzed.

While the primary use of the requirements in this element is most likely to be for nuclear power plants, the requirements are applicable to any type of radioactive material released to the atmosphere for which the release characteristics can be defined. It is recognized, however, that there may be specific applications where the source term phenomenology and atmospheric dispersion are complex. Examples of potential analyses may include the following:

(a) releases of dense and/or reactive gases (e.g., UF_6) that can have complex release and transport characteristics;

(b) releases of tritium or carbon-14, which behave differently in the environment (e.g., deposition followed by re-emission); or

(c) energetic releases (i.e., explosions where momentum effects might be significant).

Although there may be available analytical tools for determining such consequences, the SRs in this Standard may not fully address such phenomenology. Section 3 of this Standard outlines a process by which the completeness of the requirements is assessed and supplemented to meet analytical requirements. This includes the selection of appropriate models. Additionally, Section 4.3.17.9 of this Standard provides peer review requirements to ensure technical adequacy of the Radiological Consequence Analysis.

For PRAs performed prior to site selection, the use of generic site and population information to support radiological consequence analysis is expected to be consistent with appropriate assumptions and generic data sources available to support, and referenced by, the PRA.

Please refer to Section 2 for definitions of the terms *event sequence*, *source term*, *mechanistic source term*, and *release category.*

RC.2.1 Additional Information for Protective Action Parameters and Other Site Data Analysis

Results of interest in a radiological consequence analysis typically involve doses received by individuals and costs associated with radiological impacts, such as remediation of contaminated land. Past consequence analyses have found that costs are generally highly correlated to the impacted population. Thus, the population distribution surrounding a site is important to the results of a Level 3 analysis.

Many nuclear facilities have a lower population locally (e.g., within 10 miles) and larger population centers in the surrounding region (e.g., within 50 miles) of the facility. The distribution of the population surrounding a facility affects the potential impacts of a radiological release, especially when combined with prevailing wind directions.

Many nuclear facilities, such as licensed commercial nuclear plants, have prepared plans for the emergency evacuation of local populations (e.g., within approximately 10 miles). These plans are based on evacuation time estimate studies that provide estimates for how quickly local persons can be evacuated should the need arise. National, state, county, and facility guidance documents and procedures also provide important inputs regarding when different protective actions should be specified (e.g., shelter in place, partial evacuation, land interdiction). These site-specific protective actions have an important impact on the potential dose and cost consequences of a release. Some hazards (e.g., hurricanes, floods) may result in unique population responses prior to a radioactive release.

Site-specific data include local and regional land characteristics and land use (e.g., fraction of land that is not water, fraction of land devoted to farming). These site-specific data are useful to more accurately model site-specific attributes that may impact the consequences.

For PRAs performed prior to site selection, the use of generic site data is expected to be consistent with appropriate assumptions and generic sources of site data available to support, and referenced by, the PRA.

RC.2.2 Additional Information for Meteorological Data

Of particular importance to consequence analyses is rainfall amount and intensity. The frequency of occurrence and intensity of rain can effect the overall dose assessment. Rainfall results in two very important phenomena: (a) It scavenges particulates and halogens out of the atmosphere, which affects inhalation doses, and (b) the radioactive material that is deposited on the ground results in radiation dose from the groundshine pathway. When radioactive material is removed from the air, the dose due to the plume shine and inhalation pathways is reduced as the distance increases from the source.

Wind direction is important when population centers, sensitive receptors, and food crop and meat animal locations are considered. If there is a higher frequency of wind blowing toward a population center or farm area, then the overall impact and risk to the population at large would be higher. These circumstances would result in larger health effects.

Wind speed is important in determining the plume dilution, as well as the transport time, which, in turn, affects shelter/evacuation decision-making. In addition, wind speed affects plume rise, as higher winds tend to limit plume rise. Wind speed also affects the atmospheric stability. Faster winds create a well-mixed condition, which is a neutral stability that can occur any time of the day. Lighter winds are more conducive to very stable conditions at night and very unstable conditions during the day.

Atmospheric stability is used to determine the horizontal and vertical turbulence intensities in the atmosphere. More turbulence during unstable conditions promotes better dispersion and lower doses. Generally speaking, there is more turbulence in the daytime than at night due to the ground heating by incoming solar radiation and subsequent convective eddy formation. When winds are strong, the effects of heating in the daytime and cooling at night are not as important, as a well-mixed condition occurs.

For PRAs performed prior to site selection, the use of generic site information to identify meteorological data is expected to be consistent with appropriate assumptions and generic sources of site data available to support, and referenced by, the PRA.

RC.2.3 Additional Information for Atmospheric Transport and Dispersion

Simulation of the atmospheric transport and dispersion usually requires the use of appropriate models. The most commonly used model used to characterize this "plume" of airborne material is referred to as the steady-state, straight-line Gaussian model. This model calculates ground-level instantaneous and time-integrated airborne concentrations in the plume. The amount of particulate material deposited on the ground is calculated using a constant deposition velocity. Its results are a function only of distance from the source. The more sophisticated models allow temporal changes in atmospheric stability, wind speed, and other variables for each successive hour of travel time. Some more complex codes also allow the wind speed and wind direction to change with time or develop three-dimensional wind fields to account for the influence of a nonuniform wind field affected by terrain obstacles or sea breeze flows. For instantaneous releases, a three-dimensional Gaussian puff model is usually employed. Generally speaking, releases of 1 min or less are considered puffs. Longer-period releases are treated with Gaussian plume models or more sophisticated models as previously discussed.

In general, consequence modeling codes simulate the fate and transport of the radioactive plume as it travels for many hours during which the meteorological conditions are very likely to change in both time and space. In principle, there will be a different sequence of hourly weather changes for each of the 8,760 h during a full year at which the release might take place. When there were slower computers, it was impractical to run each of these sequences in turn, and some statistical method was devised for selecting a random sample by selecting starting times that are equally spaced throughout the year or by first combining the weather sequences into groups in which the pattern of hourly weather changes is similar (e.g., joint frequency distributions) and then ensuring that the sampling process covers all of the groups without bias. The question of how best to sample weather data is important. Contemporary computing techniques are now capable of running all hours separately. In this manner, the very-low-probability "tails" of the distribution can be determined for consideration in the analysis.

The Gaussian model can be modified to take into account a number of phenomena, although by its steady-state assumption, it is limited in describing certain highly complex atmospheric phenomena (e.g., airflow trajectory reversals). Allowance is usually made for the mixing of the radioactive plume as it emerges into the turbulent wake due to the aerodynamic effects on the wind field by a nearby building. The planetary boundary layer, which is the layer of turbulent air adjacent to the surface of the earth, is almost always capped by an overhead inversion, which is a layer of very stable air that acts as an effective barrier to the upward dispersion of the plume. The height of the base of this layer, often termed the "mixing height," depends on several phenomena, including the intensity of turbulence in the layer of air beneath it, which, in turn, depends on the time of day and the wind speed. Mixing heights are generally lower at night when inversions occur.

If the release scenario involves a heated discharge, the plume is buoyant due to the temperature difference between the plume and the ambient air, and it will rise according to plume-rise algorithms. The plume will also rise due to the momentum associated with the exit velocity. When there are strong winds, the vertical rise of the plume is limited, and it assumes a more horizontal path, while during calm wind conditions, the plume rises straight up until reaching equilibrium with the atmosphere. Some codes allow the plume to penetrate the inversion lid, although most reflect the plume back to the ground.

As the plume of radioactive material travels downwind from the source location, various mechanisms remove the airborne material. In addition to radioactive decay, which is dependent only on plume travel time and is a function of only the wind speed, the radioactive material is also removed (i.e., depleted) by dry deposition, due to settling, and by precipitation scavenging or wet deposition. The rate of precipitation, chemical form of the radioactive material, particle density and size distribution, surface characteristics of the ground, and meteorological conditions all affect the deposition processes. Wet deposition is characterized by a simple exponential removal rate, which is dependent on the rate of precipitation. When the occurrence of precipitation is specified by the weather data, it is assumed to occur uniformly with time and throughout the spatial interval in which the plume is located. Plumes may also lose material if they impact on vegetation or terrain surfaces before reaching the ground.

Noble gases are assumed to be insoluble and nonreactive and therefore are not removed by wet deposition. Since gases do not have a fall velocity, but remain within the turbulent flows of the atmosphere, they are not removed by dry deposition.

For PRAs performed prior to site selection, the use of site data to characterize the atmospheric transport and dispersion conditions is expected to be consistent with appropriate assumptions and generic sources of site data available to support, and referenced by, the PRA.

RC.2.4 Additional Information for Dosimetry

The dosimetry model includes the appropriate pathways contributing dose to individual receptors and population groups over short-term and long-term exposures. Exposure pathways are associated with the passing plume and ground contamination resulting from deposition of radionuclides, as well as subsequent resuspension of deposited material and ingestion of contaminated food and water.

Radiological exposures in a radiological consequence analysis account for both short-term and long-term effects. The short term considers plume passage and a limited time afterward (on the order of days). The long term considers indirect uptake of radioactivity over an extended period of time (on the order of years).

The pathways of exposure include the following:

(a) direct external exposure to radioactive material in the plume [principally due to gamma radiation (cloudshine)];

(b) exposure from inhalation of radionuclides in the cloud and resuspended material deposited on the ground;

(c) exposure to radioactive material deposited on the ground (groundshine);

(d) uptake of radioactive material deposited onto the body surfaces (skin deposition);

(e) ingestion from deposited radionuclides that make their way into the food and water pathways.

Dosimetry may include consideration of protective actions to limit dose. This consideration is often in the form of shielding or protection factors. Mitigation actions are addressed in the Protective Action Parameters and Other Site Data Analysis subelement of this Standard.

RC.2.4.1 Dosimetry Basis Model

Dosimetry models used in the radiological consequence analysis should comply with current models and associated parameters accepted by the international community, such as the International Commission on Radiological Protection.

RC.2.4.2 Dose Conversion Factors

The dose received from radioactive material is specific to an organ or tissue and is estimated by a DCF. The DCFs take into account the migration of the radioisotope within the body, the decay of the radioisotope, and the formation of daughter isotopes that may be radioactive.

The DCF values are typically based on exposure to an adult assuming a particle size of 1.0 µm activity mean aerodynamic diameter. These values are generally applied uniformly for all ages in the general public under all release conditions.

RC.2.4.3 Consumption Pathways

Deposition from an airborne plume may contaminate water and food supplies. The uptake of radionuclides by plants and animals, and their transfer into the food chain for humans, is a very complex process.

Consumption of contaminated food products is not restricted to persons living near the site of an accidental release since the food products may be transported to another location for processing and consumed in still another location. The ingestion dose should therefore be calculated separately from the other doses (from inhalation, etc.). It is not to be added to the doses from the other modes of intake unless it is clear that the receptor for the ingestion dose is the same as the receptor for the other modes of intake. This is important if only a portion of the total dose is to be used for this purpose [e.g., dose to the population within 80 km (50 miles) of the site for cost-benefit applications]. If the analysis uses total dose and a linear, no-threshold dose response model, then the food pathway can be added to the other pathways without biasing the result. Once the amount of radioactive material ingested has been determined, the dose can be calculated by multiplying this amount by the DCF for ingestion.

When radioactive material is deposited on the ground through dry and/or wet deposition, some fraction of this material may eventually be transported into the potable water consumed by humans. This can be through (a) direct deposition to surface bodies of water and uptake through the drinking water supply, or (b) deposition to land surfaces with subsequent transfer to potable water supplies through wash off.

RC.2.4.4 Cloudshine and Groundshine

Cloudshine doses are primarily from gamma and beta radiation emitted from a plume during its passage. Simple cloudshine models are better termed as immersion models and do not account for any spatial variation in concentration. True cloudshine models account for the dimensions of the plume and the relative location of the receptor. In addition, buildings and other structures may offer protection from cloudshine in terms of shielding.

The treatment of groundshine is similar to that of cloudshine. The amount of gamma radiation received by a receptor depends on the concentration of a specific isotope on the ground. Most groundshine models assume that the receptor is standing on a planar surface with a uniform radionuclide concentration. Groundshine can continue over an extended period, so the exposure period chosen by the analyst can be an important consideration.

RC.2.4.5 Skin Deposition

Doses from skin deposition are relatively small and of short duration (a few hours). The primary radionuclides of importance for skin contamination are the beta emitters. Beta particles can penetrate the surface layer of dead skin cells and damage the cells directly beneath. The dose is integrated over the time duration that the material is on the skin prior to decontamination to give the skin DCF.

For PRAs performed prior to site selection, the use of population data to support dose calculations is expected to be consistent with appropriate assumptions and generic sources of population data available to support, and referenced by, the PRA.

RC.2.5 Additional Information for Health Effects

Health effects from exposure to ionizing radiation are usually divided into two categories depending on the dose received and the dose rate:

(a) Nonstochastic or deterministic health effects, also called early (or prompt) effects, caused by doses exceeding certain thresholds. These health effects include both mortality and morbidity (i.e., fatalities and injuries) as outcomes and typically occur within the first few days or weeks following the exposure;

(b) Stochastic or latent health effects, which may occur several years after exposure. The latent health effects also include mortality and morbidity as outcomes. Latent health effects are usually modeled with a linear, no-threshold dose-response relationship, although some codes contain other (e.g., linear quadratic) response functions and may also include provisions to include a user-defined threshold for cancer induction.

The health effects caused by radiation exposure are subject to considerable uncertainty, and the models used to relate dose and response should incorporate and, to the extent possible, quantify this uncertainty. The uncertainty can be subdivided further into parameter uncertainty and model uncertainty. Parameter uncertainty arises partly from the random or stochastic nature of the process of cell damage caused by radiation and partly from the inherent error involved in drawing inferences of effects based on small samples. Parameter uncertainty is typically characterized by establishing a probability distribution on the parameter values. This distribution expresses an analyst's degree of belief in the values the parameters could take, based on the data available. Model uncertainty is more difficult to estimate since it arises from physical limitations, such as the need to rely on analogies from animal data in estimating, for example, the risk of pulmonary syndrome mortality. Also, estimates of radiation-induced cancers rely largely on the extrapolation of Japanese atomic bomb survivor data from the high dose and high dose rates received by survivors to the low-dose, low-dose-rate region.

Early fatality and early injury health effects are generally modeled using a cumulative hazard function with a threshold, and a number of sigmoidal functions, such as the Weibull, probit, and logistic functions, can be used. One approach in some codes is based on the Weibull hazard function. If the dose is less than the threshold dose for that particular organ and health effect, then the risk for that is set to zero. Incorporation of dose rate effects that account for the reduction in health effects of dose protraction are accomplished by suitably adjusting the value of the dose used in the hazard function over the various time intervals of interest.

Early health effects from radiation exposure that are generally considered to lead to mortality include the following syndromes and target organs:

(a) hematopoietic syndrome—the killing of blood cell precursors in the marrow after irradiation with the target organ being the red bone marrow;

(b) pulmonary syndrome—damage to the lungs as the target organ;

(c) gastrointestinal syndrome—damage to the small intestine and the colon as the target organs.

Early health effects that are considered to lead to morbidity (injury) include the following:

(a) prodromal syndrome—gastrointestinal and neurovascular symptoms;

(b) radiation pneumonitis— lung impairment;

(c) hypothyroidism—thyroid organ impairment;

(d) skin burn—skin erythema caused by radiation injury to the basal cells below the skin surface.

Other early health effects from radiation exposure include impacts on the reproductive system, including the ovaries and testes, and effects on the embryo and fetus from irradiation that may include fetal death and mental retardation.

Latent health effects, mainly cancers, are most often modeled via a linear or linear quadratic relationship between dose and response. There is considerable scientific debate regarding the presence or absence of a threshold in the dose-response relationship used to model cancer incidence following irradiation. The latest position of the national and international bodies concerned with radiation protection, as expressed in BEIR VII [RC-10] and ICRP Publication 103 [RC-11], affirm the no-threshold hypothesis. Some computer codes include provisions for a user-defined threshold that could be employed for certain purposes as an alternative method to calculate latent cancer fatalities. If a threshold is used, the value should be justified for the intended purpose. The risk coefficient relating risk of health effect to dose in the linear model can be modified to incorporate the effects of higher dose and of lower dose rate.

Latent health effects from radiation exposure include both mortality and morbidity as outcomes. Leukemia and bone cancer are generally modeled as fatalities. Most of the remaining latent health effects, cancers of the lung, breast, gastrointestinal tract, thyroid, and bladder, can be modeled with different risk coefficients for either mortality or morbidity as outcomes. Skin cancer is usually modeled only as leading to morbidity. Latent health effects may also include childhood cancers from exposures in utero and genetic effects that could lead to an increase in birth defects among the children of the exposed population.

The health effects discussed in this Standard have been limited to human populations.

RC.2.6 Additional Information for Economic Factors

The economic factors that enter into an off-site consequence analysis following a radiological release are those related to the economic impacts of the release on the surrounding land and the population. These factors include the costs of various actions (e.g., evacuation, relocation, decontamination) taken to protect the public from short-term and long-term exposure via different exposure pathways, the costs of health effects and health care following exposure, and secondary economic effects.

Short-term evacuation costs include costs related to transport, food, housing, and, possibly, lost income for the time period that the affected population remains evacuated. It is evaluated in dollars per person per day. These costs can vary considerably by state and region. Similarly, short-term or temporary relocation costs may be incurred as a protective measure for people who may not have been evacuated initially in the emergency phase or may have had to extend their initial evacuation period. These costs depend on the period of time the affected population remains relocated and are similar to those for evacuation and measured in the same units.

To protect against possible ingestion doses, agricultural products such as crops, dairy products, etc., that may have been contaminated by fallout from the release may need to be disposed of. The cost of crop disposal is estimated from the fraction of the region that is farmland, the extent of area affected where doses from ingestion would exceed acceptable limits, the average annual farm production per unit area, and whether the release occurs during the growing season or not. Release events that occur outside the growing season may not incur any crop disposal costs. Milk and dairy disposal costs consider the fraction of farm sales that are specifically dairy products and also the time for radioactive levels in milk to reach levels acceptable for ingestion. Many of these costs may be very site-specific and depend on the value of farm production in the area, the cost of land and farm improvements, etc.

Long-term protective actions include relocation (temporary or permanent) of people and businesses from contaminated areas that have been rendered uninhabitable, decontamination, and interdiction of contaminated land and property (temporary or permanent). Each of these actions involves costs to society. Relocation costs for people and businesses that may have to remain relocated for fairly long periods of time such as a few years in a region rendered uninhabitable are expressed in dollars per person. These costs measure both personal and business losses for a period of transition and may include moving expenses. Decontamination costs depend on the actions taken during the long term to reduce doses to acceptable levels. Several levels of decontamination may be defined in terms of increasing effectiveness and cost, where effectiveness is measured by reduction of projected dose. Decontamination costs can be defined separately for farmland and non-farmland areas and evaluated in dollars per unit area for farmland and dollars per person for non-farmland areas. If the maximum level of decontamination is not able to reduce projected doses to an acceptable level within a user-defined period, then the land or property may be permanently condemned.

Several approaches may be employed to determine the economic impact of long-term interdiction or permanent condemnation of land areas. Interdictions imply a disturbance such as loss of productivity and, more generally, loss of income and wealth in the local and regional economies. These approaches include estimation of the rate of output of land and all other productive assets in the area and integration of this value over the interdiction period. A second approach uses the concept of the wealth of a particular region to estimate the total present value of land and other assets in the affected area. A third approach uses economic input-output modeling techniques applied at a regional level to estimate economic losses over a period. Many of these costs, such as regional or state wealth or productivity, are also site-specific.

The costs of health effects are typically estimated by two approaches: national output maximization and social welfare maximization. In the former approach, the cost of the health effect is estimated by the discounted present value of the loss of the person's future earnings (or output) due to the incident. Allowances are made for nonmarketed output (e.g., services of home care providers) and other costs, such as medical and legal expenses, as well as ad hoc factors to deal with "pain and suffering." In the latter approach, individual willingness to pay for safety is estimated and then aggregated over all affected individuals.

Secondary impacts of release costs include several factors, such as an increase in the cost of electricity that produces ripple effects in a wider region and population redistribution from permanent relocation, which affects employment, incomes, and productivity.

For PRAs performed prior to site selection, the use of generic cost parameter data and generic site data is expected to be consistent with appropriate assumptions and generic sources of data available to support, and referenced by, the PRA.

RC.2.7 Additional Information for Consequence Quantification

Radiological consequence analyses are generally performed using codes specifically designed for this purpose. While many different codes have been developed and used worldwide in the last 30 yr, relatively few radiological consequence analysis codes are currently supported. Reference [RC-1] contains a brief overview of known computer codes. These codes model the consequences associated with a postulated release such that the code results produced are conditional. Assessment of risk requires the integration of Radiological Consequence Analysis conditional results with Event Sequence Analysis and Mechanistic Source Term Analysis results (e.g., release frequencies). Such integration is addressed in Section 4.3.16 and Section 4.3.17.1 of this Standard.

Each radiological consequence analysis code includes algorithms that have calculation limitations. The radiological consequence analyst must ensure that modeling is appropriately performed within the range of applicability of the code. Such applicability is influenced not only by calculation limitations, but also by the outputs of interest. For example, mean regional results (e.g., 50-mile-radius population dose) are generally less sensitive to terrain impacts than results for a particular location. Therefore, the use of a radiological consequence analysis code for site surrounded by varied terrain may be acceptable for a regional analysis but may not be acceptable for emergency response decision-making near the site.

Radiological consequence analysis results are reviewed to confirm proper code execution and that the results are reasonable. Risk-significant contributors to results of interest are identified and uncertainties assessed. The quantification process and results are be documented in a manner that facilitates applications, upgrades, and peer review. Results of interest may include mean values for consequences of interest (e.g., 50-mile population dose, 50-mile economic cost, early fatalities), upper-bound values based on weather variability (e.g., 95th percentile), and complementary cumulative distribution function (CCDF) results for particular metrics to demonstrate the pairing of consequence and probability based on weather variability.

4.3.18 Risk Integration (RI)

This Section presents the technical requirements associated with Risk Integration.

4.3.18.1 Objectives and Technical Requirements for Risk Integration

The objectives of the Risk Integration element ensure that

(a) The criteria for establishing the appropriate absolute or relative risk significance of event sequences, event sequence families, structures, systems and components (SSCs), and basic events modeled in the PRA are defined and justified;

(b) The overall risk is calculated, and the significant contributors to risk are identified using appropriate risk metrics. Risks from each reactor and radionuclide source and combination of reactors and sources are included in Risk Integration;

(c) The uncertainties associated with the risk results are characterized and calculated, and the major contributors to risk uncertainties are identified;

(d) The Risk Integration is documented to provide traceability of the work.

Table 4.3.18.1-1 High Level Requirements for Risk Integration

Designator	Requirement
HLR-RI-A	The Risk Integration analysis shall define the criteria to be used to establish the risk significance of event sequences, event sequences families, systems, structures and components, and basic events modeled in the PRA. These criteria shall be defined in a manner that is consistent with the intended applications of the PRA.
HLR-RI-B	The Risk Integration analysis shall calculate the overall risk and identify the significant risk contributors using risk metrics consistent with the selected risk significance criteria and appropriate for the intended PRA applications. The Risk Integration analysis shall include the integrated risk including those from each reactor and radionuclide source and combination of reactors and sources.
HLR-RI-C	The Risk Integration shall characterize and quantify the uncertainties in the calculated risk metrics to the extent practical. Key sources of model uncertainty and assumptions shall be identified, and their potential impact on the results shall be understood.
HLR-RI-D	The documentation of the Risk Integration shall provide traceability of the work.

Table 4.3.18.1-2 Supporting Requirements for HLR-RI-A

The Risk Integration analysis shall define the criteria to be used to establish the risk significance of event sequences, event sequences families, systems, structures and components, and basic events modeled in the PRA. These criteria shall be defined in a manner that is consistent with the intended applications of the PRA. (HLR-RI-A)

Index No. RI-A	Capability Category I	Capability Category II
RI-A1	For the purpose of Risk Integration, DEFINE the consequence measure(s) necessary to support the intended applications of the PRA (e.g., person-rem, early fatalities, latent health effects, site boundary doses, quantity of radioactive material released).	
RI-A2	For PRA applications involving the calculation of baseline risk, DEFINE and JUSTIFY the criteria to be used to establish the relative risk significance of PRA model elements that accounts for both the frequency and consequences of modeled event sequences. USE the risk significance criteria in Table 1.9-1 or JUSTIFY alternate criteria. See Note RI-N-1	
RI-A3	For PRA applications in which the acceptability of the risk against fixed risk targets is considered, DEFINE and JUSTIFY the criteria to be used to establish the absolute risk significance of PRA model elements that accounts for both the frequency and consequences of modeled event sequences. USE the risk significance criteria in Table 1.9-2 or JUSTIFY alternate criteria. See Note RI-N-1, RI-N-2	
RI-A4	USE a minimum reporting frequency of 10^{-7} per plant-year for all modeled event sequence families or JUSTIFY an alternative. See Note RI-N-3	
RI-A5	USE a minimum reporting consequence of 10% of the consequences due to background radiation dose for consequence metrics associated with all modeled event sequence families or JUSTIFY an alternative. See Note RI-N-4	

Table 4.3.18.1-3 Supporting Requirements for HLR-RI-B

The Risk Integration analysis shall calculate the overall risk and identify the significant risk contributors using risk metrics consistent with the selected risk significance criteria and appropriate for the intended PRA applications. The Risk Integration analysis shall include the integrated risk including those from each reactor and radionuclide source and combination of reactors and sources. (HLR-RI-B)

Index No. RI-B	Capability Category I	Capability Category II
RI-B1	COMPILE information on event sequences families and their frequencies from SRs of HLR-ESQ-D and the associated consequences from SRs of HLR-RCQ-D.	
RI-B2	CALCULATE the integrated risk results using risk metrics for the PRA defined in the Supporting Requirements (SRs) of HLR-RI-A and the event sequence families compiled in Requirement RI-B1. USE at least one of the following approaches for calculating the integrated risk metrics or JUSTIFY an alternative: *(a)* sum of the product of the point estimate frequencies and point estimate consequences of each modeled event sequence, event sequence family, or release category; *(b)* plots of point estimate frequencies and consequences of individual event sequences, event sequence families, or release categories on a frequency-consequence plot in comparison with frequency-consequence targets selected for PRA applications. See Note RI-N-5	QUANTIFY the integrated risk results using risk metrics for the PRA defined in the SRs of HLR-RI-A and the event sequence families defined in Requirement RI-B1. USE the following approaches for quantifying the integrated risk metrics or JUSTIFY an alternative: *(a)* sum of the product of the mean frequencies and mean consequences of each modeled event sequence, event sequence family, or release category; *(b)* plots of mean frequencies and consequences and associated uncertainties of individual event sequences or event sequence families on a frequency-consequence plot in comparison with frequency-consequence targets selected for PRA applications; *(c)* exceedance frequency [i.e., complementary cumulative distribution function (CCDF)] curves that show the frequency of exceedance of selected consequence metrics (e.g., radioactive material release limits, site-boundary dose, person-rem, early fatalities, latent cancer fatalities). See Note RI-N-5
RI-B3	When aggregating the results from different sources of radioactive material, hazard groups (e.g., internal events, internal floods, internal fires, seismic events), and plant operating states, IDENTIFY contributions from different sources of radioactive material and hazard group to each risk-significant event sequence family frequency. See Note RI-N-5	When aggregating the results from different sources of radioactive material, hazard groups (e.g., internal events, internal floods, internal fires, seismic events), and plant operating states, IDENTIFY possible differences in the level of detail and degree of conservatism and realism in the modeling of event sequences for different sources of radioactive material and in different hazard groups. REVIEW the contributions to the event sequence family frequencies and the determination of risk-significant event sequences and basic events separately for each source of radioactive material and hazard group, as well as for any aggregated results.
RI-B4	INCLUDE the risk contributions from modeled event sequence families involving releases from multiple reactors and unscreened sources of radioactive material included in the scope of the PRA.	

Table 4.3.18.1-3 Supporting Requirements for HLR-RI-B (Cont'd)

The Risk Integration analysis shall calculate the overall risk and identify the significant risk contributors using risk metrics consistent with the selected risk significance criteria and appropriate for the intended PRA applications. The Risk Integration analysis shall include the integrated risk including those from each reactor and radionuclide source and combination of reactors and sources. (HLR-RI-B)

Index No. RI-B	Capability Category I	Capability Category II
RI-B5	JUSTIFY that selection of the event sequence variations within each event sequence family are not risk-significant, including both the frequency and release magnitude in the characterization of risk significance for this purpose. ENSURE that both the selection of release categories and the assignments of event sequences to families are sufficient to avoid grouping non-risk-significant contributors with risk-significant contributors for each release category. See Note RI-N-6	
RI-B6	IDENTIFY risk-significant contributions in a manner that is sufficient to derive insights within the scope and level of detail of the PRA models. See Note RI-N-7	
RI-B7	USE methods and codes for the Risk Integration that account for method and code limitations. JUSTIFY application of these methods and codes for the scope of hazards, plant operating states, release categories, and event sequences within the scope of the PRA.	

Table 4.3.18.1-4 Supporting Requirements for HLR-RI-C

The Risk Integration analysis shall characterize and quantify the uncertainties in the calculated risk metrics to the extent practical. Key sources of model uncertainty and assumptions shall be identified, and their potential impact on the results shall be understood. (HLR-RI-C)

Index No. RI-C	Capability Category I	Capability Category II
RI-C1	COMPILE a list of key sources of model uncertainty and related assumptions identified in each technical element of this Standard along with the associated assessment of the impact of that uncertainty in both event sequence family frequencies (see Requirement ESQ-E1) and consequences (see either Requirement MS-D3 or RCQ-C1, as applicable). INCLUDE screening out items from the PRA model including hazard groups, hazard events, plant operating states, initiating events, event sequences, and basic events. See Note RI-N-5, RI-N-8	
RI-C2	REVIEW the grouping of event sequences into event sequence families and ENSURE that the uncertainty associated with the grouping does not artificially cause the event sequence families to be risk-significant. See Note RI-N-5	
RI-C3	ASSESS the effects of the items compiled in Requirement RI-C1 by performing a qualitative or quantitative evaluation of the effects of the individual sources of uncertainty or of combinations of interest on each modeled risk metric. See Note RI-N-5	

Table 4.3.18.1-4 Supporting Requirements for HLR-RI-C (Cont'd)

The Risk Integration analysis shall characterize and quantify the uncertainties in the calculated risk metrics to the extent practical. Key sources of model uncertainty and assumptions shall be identified, and their potential impact on the results shall be understood. (HLR-RI-C)

Index No. RI-C	Capability Category I	Capability Category II
RI-C4	CHARACTERIZE the uncertainty distribution of the selected risk metric results by specifying or discussing the range of the uncertainty, consistent with the characterization of parameter uncertainties. See Requirements ESQ-E2 and either MS-D4 or RCQ-C2, as applicable. See Note RI-N-5, RI-N-9	CALCULATE the uncertainty distribution of the selected risk metric results by propagating the uncertainty distributions on the parameters for the risk-significant contributors, taking into account those model uncertainties explicitly characterized by a probability distribution. See Requirements ESQ-E2 and either MS-D4 or RCQ-C2, as applicable. CALCULATE the uncertainties in such a way that either the state-of-knowledge correlation between event frequencies or the dependencies between uncertain phenomena is taken into account when either the state-of-knowledge correlation between component failure basic event probabilities or dependencies between uncertain phenomena are risk-significant. See Note RI-N-5, RI-N-9

Table 4.3.18.1-5 Supporting Requirements for HLR-RI-D

The documentation of the Risk Integration shall provide traceability of the work. (HLR-RI-D)

Index No. RI-D	Capability Category I	Capability Category II
RI-D1	DOCUMENT the process used in the Risk Integration, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* criteria used to define either the absolute or relative risk significance of event sequences, event sequence families, release categories, plant operating states, hazard groups, and basic events in the PRA model; *(b)* sufficient information to understand the results of the risk assessment, the risk-significant contributions, and risk insights necessary to support intended applications; *(c)* limitations in the results due to limitations or omissions from the PRA scope; *(d)* traceability to the risk-significant contributions to the results of the Event Sequence Quantification, mechanistic source term evaluation, and Radiological Consequence Analysis; *(e)* results of comparison of risk results to risk significance and acceptance criteria as applicable; *(f)* key sources of uncertainty in the results; *(g)* risk insights regarding reactor design features that help to explain the risk results; *(h)* risk contributors from basic events, human failure events (HFEs), SSCs, plant operating states, hazard groups, and sources identified in Requirement ESQ-F2 that make significant contributions to the total integrated risk.	
RI-D2	DOCUMENT the sources of model uncertainty, related assumptions, and reasonable alternatives (as identified in HLR-RI-A, HLR-RI-B, and HLR-RI-C) associated with the Risk Integration.	

4.3.18.2 Peer Review Requirements for Risk Integration

4.3.18.2.1 Purpose

This Section provides requirements for peer review of the Risk Integration element of the PRA.

4.3.18.2.2 Peer Review Team Composition and Personnel Qualification

In addition to the general requirements in Section 6, the peer review team shall have collective knowledge and experience in the area of risk integration. The team members assigned to review Risk Integration shall overlap that assigned to review Event Sequence Analysis, Event Sequence Quantification, Mechanistic Source Term Analysis, and Radiological Consequence Analysis to ensure consistency between the modeling for these elements. The team members assigned to review the Risk Integration shall have experience specific to these areas and the capability of recognizing plant and design-specific features of the analyses.

4.3.18.2.3 Review of Risk Integration to Confirm Results

The scope of the review of the Risk Integration includes the following:

(a) the selection of absolute or absolute risk significance criteria as appropriate for the PRA application;

(b) justification for use of selection of risk significance criteria;

(c) the selection of risk metrics and consequence metrics for the Risk Integration;

(d) the summary and interpretation of results for the total integrated risk and risk for specific combinations of sources, plant operating states, and hazards;

(e) the evaluation of aggregation issues when risk is aggregated across different sources, plant operating states, and hazard groups as required in SR RI-B3;

(f) the evaluation of the risk-significant event sequences, event sequence families, and release categories that are determined in the Risk Integration element;

(g) the identification and evaluation of uncertainty in the integrated risk results.

4.3.18.3 References for Risk Integration

The following is a list of publications referenced in this Standard.

[RI-1] SC-29980-203, "Fluoride-Cooled High Temperature Reactor Licensing Modernization Project Demonstration," Southern Company, September 2019

[RI-2] SC-29980-201, "PRISM Reactor Licensing Modernization Project Demonstration," Southern Company, December 2018

[RI-3] SC-29980-202, "Westinghouse eVinci™ Micro-Reactor Licensing Modernization Project Demonstration," Southern Company, August 2019

[RI-4] SC-29980-200, "High Temperature, Gas-Cooled Pebble Bed Reactor Licensing Modernization Project Demonstration," Southern Company, August 2018

[RI-5] EPRI AR LR 2019-06, "Molten Salt Reactor Experiment Using Risk-Informed, Performance-Based Technical Guidance to Inform Future Licensing for Advanced Non-Light Water Reactors," Electric Power Research Institute, September 2019

[RI-6] NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with Probabilistic Risk Assessment in Risk-Informed Decision Making (RIDM)," U.S. Nuclear Regulatory Commission, March 2009

[RI-7] EPRI 1016737, "Treatment of Parameter and Modeling Uncertainty for Probabilistic Risk Assessments," Electric Power Research Institute, December 2008

[RI-8] EPRI 1009652, "Guideline for the Treatment of Uncertainty in Risk-Informed Applications: Technical Basis Document," Electric Power Research Institute, December 2004

[RI-9] EPRI 1026511, "Practical Guidance on the Use of Probabilistic Risk Assessment in Risk-Informed Applications with a Focus on the Treatment of Uncertainty," Electric Power Research Institute, December 2012

[RI-10] EPRI TR-3002003116, "An Approach to Risk Aggregation for Risk-Informed Decision-Making," Electric Power Research Institute, Palo Alto, CA, 2015

[RI-11] EPRI 3002014783, "A Framework for Using Risk Insights in Integrated Risk-Informed Decision-Making," Electric Power Research Institute, Palo Alto, CA, 2019

[RI-12] NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing," U.S. Nuclear Regulatory Commission, December 2007

[RI-13] ANSI/ANS-53.1-2011 (R2016), "Nuclear Safety Design Process for Modular Helium-Cooled Reactor Plants," American Nuclear Society, 2011

[RI-14] NEI 18-04 (Rev. 1), "Risk-Informed Performance-Based Technology Inclusive Guidance for Advanced Reactor Licensing Basis Development," Nuclear Energy Institute, August 2019

[RI-15] Regulatory Guide 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance Based Approach to Inform the Content of Applications for Licenses, Certifications, and Approvals For Non-Light-Water Reactors," U.S. Nuclear Regulatory Commission, June 2020

[RI-16] "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement," Federal Register, Vol. 51, No. 149, Office of the Federal Register (August 4, 1986); republished with corrections August 21, 1986, Federal Register, Vol. 51, No. 160

[RI-17] G. APOSTOLAKIS and S. KAPLAN, "Pitfalls in Risk Calculations," Reliab. Eng., 2,135, 1981

[RI-18] Wash-1400, "The Reactor Safety Study," (also known as NUREG-75/014); Publisher: U.S. Nuclear Regulatory Commission (NRC), 11555 Rockville Pike, Rockville, MD, 20852, 1975

NONMANDATORY APPENDIX RI: NOTES AND EXPLANATORY MATERIAL FOR RISK INTEGRATION

RI.1 NOTES ASSOCIATED WITH RISK INTEGRATION

Table RI-1 Notes Supporting Risk Integration Requirements

Number	Notes
RI-N-1	Depending on the application and scope of the PRA, only one of these SRs (RI-A2 or RI-A3) may apply. See RI-A2, RI-A3
RI-N-2	Acceptable absolute risk significance approaches to event sequence family and SSC risk significance were demonstrated in the Licensing Modernization Project's demonstration reports. See [RI-1], [RI-2], [RI-3], [RI-4], and [RI-5] See RI-A3
RI-N-3	The purpose of the minimum reporting frequency is to recognize the limitations of PRA technology in the estimation of the frequency of rare events. Calculated event sequence and supporting cutset frequencies that are below this value are often the result of unprovable assumptions about the independence of the underlying basic events. There are natural phenomena such as major asteroid or meteor impacts that would result in life extinction on the earth at frequencies on the order of 10^{-7} per year. Hence the completeness of the selection of initiating events and event sequences cannot be assured at such low frequencies. In addition, the screening criteria SCR-1 permits the screening out of event sequence families with frequencies less than 10^{-7} per year. When event sequences or event sequence families are calculated in the PRA model to have frequencies less than the minimum reporting value, the results should be reported as simply < [minimum reporting value]. See RI-A4
RI-N-4	The population that may be exposed to a radiation dose from a release from the plant will also be exposed to background radiation during the release and source term exposure. In the U.S., the average background radiation dose is approximately 300 mrem Total Effective Dose Equivalent (TEDE) per year. If there is a release that occurs over a 30-day period, a person in the area around the plant would receive about 25 mrem from background radiation. In other countries and regions, the background radiation dose may be significantly higher. The purpose of this minimum reporting consequence level is to identify source term exposures less than that from background radiation to be less than minimum reporting levels. When the consequences of event sequences or event sequence families are calculated in the PRA model to be less than the reporting threshold value, the results should be reported as simply < [minimum reporting value]. See RI-A5
RI-N-5	The following documents provide a guide to U.S. light water reactor (LWR) risk-informed licensing applications, treatment of uncertainties, and risk aggregation to LWR applications which, while not directly applicable to non-LWRs, may provide useful guidance and best practices beyond their intended audiences. See [RI-6], [RI-7], [RI-8], [RI-9], [RI-10], and [RI-11] See RI-B2, RI-B3, RI-C1, RI-C2, RI-C3, RI-C4
RI-N-6	This SR is focused on preventing contributors (e.g., human actions, SSC failures) from non-risk-significant event sequences from being labeled as risk-significant simply because that non-risk-significant event sequence was grouped with risk-significant event sequences into a risk-significant event sequence family. See RI-B5

Table RI-1 Notes Supporting Risk Integration Requirements (Cont'd)

Number	Notes
RI-N-7	Acceptable approaches to analyze the results to support the identification of risk-significant contributions in a manner that is sufficient to derive insights within the scope and level of detail of the PRA models include the following: *(a)* risk-significant contributions by plant operating state, hazard group, and initiating event; *(b)* risk-significant contributions by release category, event sequence families, and individual event sequences; *(c)* risk-significant contributions by failure cause (e.g., component failures, common-cause failures, human errors, maintenance unavailability); *(d)* risk-significant functions, basic events, and basic event categories as determined by risk significance; *(e)* other well-defined modeling items of the PRA model. See RI-B6
RI-N-8	If the PRA's consequence metric is measured as releases from the building instead of off-site consequence, MS-D3 provides the appropriate information to support this SR. If the PRA's consequence metric is measured in off-site consequence, then the information from MS-D3 will be input into RCQ-C1 through RCRE-A2 and thus RCQ-C1 provides the appropriate information to support this SR. See RI-C1
RI-N-9	If the PRA's consequence metric is measured as releases from the building instead of off-site consequence, MS-D4 provides the appropriate information to support this SR. If the PRA's consequence metric is measured in off-site consequence, then the information from MS-D4 will be input into RCQ-C2 through RCRE-A2 and thus RCQ-C2 provides the appropriate information to support this SR. See RI-C4

RI.2 AN EXAMPLE OF RISK INTEGRATION USING RELATIVE SIGNIFICANCE

The purpose of this nonmandatory appendix is to provide guidance on alternative ways to characterize risk and relative risk significance for PRAs that are developed to meet the technical requirements identified in SR RI-B3 and SR RI-B4 for identifying the relative and absolute contributions to risk consistent with the risk significance criteria identified in SR RI-A2 and SR RI-A3.

Consider an example PRA that produced a set of four event sequence families, referred to here as LBE-1, LBE-2, LBE-3, and LBE-4. Each family is characterized by estimates of the frequency and exclusion area boundary (EAB) dose. These estimates include mean values as well as fully quantified uncertainties as shown in the following table. The dose uncertainties are assumed to account for uncertainties in the mechanistic source terms and the variability in the meteorological conditions meeting the associated technical requirements for Mechanistic Source Term Analysis and Radiological Consequence Analysis, respectively. Lognormal distributions are used in this example for both frequency and consequence estimates, and these are characterized in terms of means and error factors.

Table RI-2 Example of Relative Risk Determination

Event Sequence Analysis Family	Frequency/Plant-Year		EAB Dose (rem)		Mean Risk (rem/plant-year)	Percent Total Risk
	Mean	EF[1]	Mean	EF[1]		
LBE-1	1.00E-01	3	0.025	5	2.50E-03	95.4%
LBE-2	1.00E-03	10	0.1	10	1.00E-04	3.8%
LBE-3	1.00E-05	30	5	20	2.00E-05	0.8%
LBE-4	1.00E-04	10	.01	10	1.00E-06	0.04%
Total Risk					2.62E-03	100%

Note [1] EF = error factors of assumed lognormal distributions for frequency and dose;
EF = SQRT (95%tile/5%tile)

There are two established approaches to characterize the results of a PRA that includes a quantification of event sequence frequencies and consequences. One is based on using mean point estimates of the frequency and consequence of each event sequence family, calculating the risk by multiplying frequency and consequence, and summing up over the event sequence families. This metric is used in SR RI-A2 to establish relative risk significance for Capability Category I (CC-I). The other approach that was used in Wash-1400 [RI-18] and in many LWR Level 3 PRAs is based on calculating the so-called "Complementary Cumulative Distribution Function," which is a curve of the frequency of exceeding a given level of consequence versus consequence level. This metric is used in SR RI-A3 to establish relative risk significance for Capability Category II (CC-II). Both approaches are applied to the above example set of event sequence families in the following:

RI.2.1 Point Estimate of Risk Method

The total baseline risk obtained using mean point estimates of the event sequence family frequencies and doses is obtained using the following Equation (1).

$$\text{Risk} = \sum_{j=1}^{4} f_j d_j = 2.6 \times 10^{-3}\,\text{rem/plant-year} \qquad \text{(RI.1)}$$

Where:

f_j is the mean frequency of event sequence family j

d_j is the mean EAB dose for event sequence family j

Assuming a relative risk significance threshold of 1% consistent with the definition of relative risk significance in Table 1.9-1, it is seen in the above table that event sequence families LBE-1 and LBE-2 would be classified as risk-significant and LBE-3 as not relative risk-significant. Actually, LBE-1 essentially dominates the calculation of risk, and LBE-2 makes a contribution that marginally exceeds the selected relative risk-significant threshold. It is noted that the above characterization of relative risk significance does not explicitly account for the uncertainty in the frequency and dose estimates except to the extent that uncertainties influence the mean values of frequency and consequence. Uncertainty in the dose estimate is explicitly addressed by the exceedance frequency method discussed in the following.

RI.2.2 Exceedance Frequency of Consequence Method

$$F(d \geq x) = \sum_{j=1}^{4} f_j P_j\,(d \geq x), \qquad \text{(RI.2)}$$

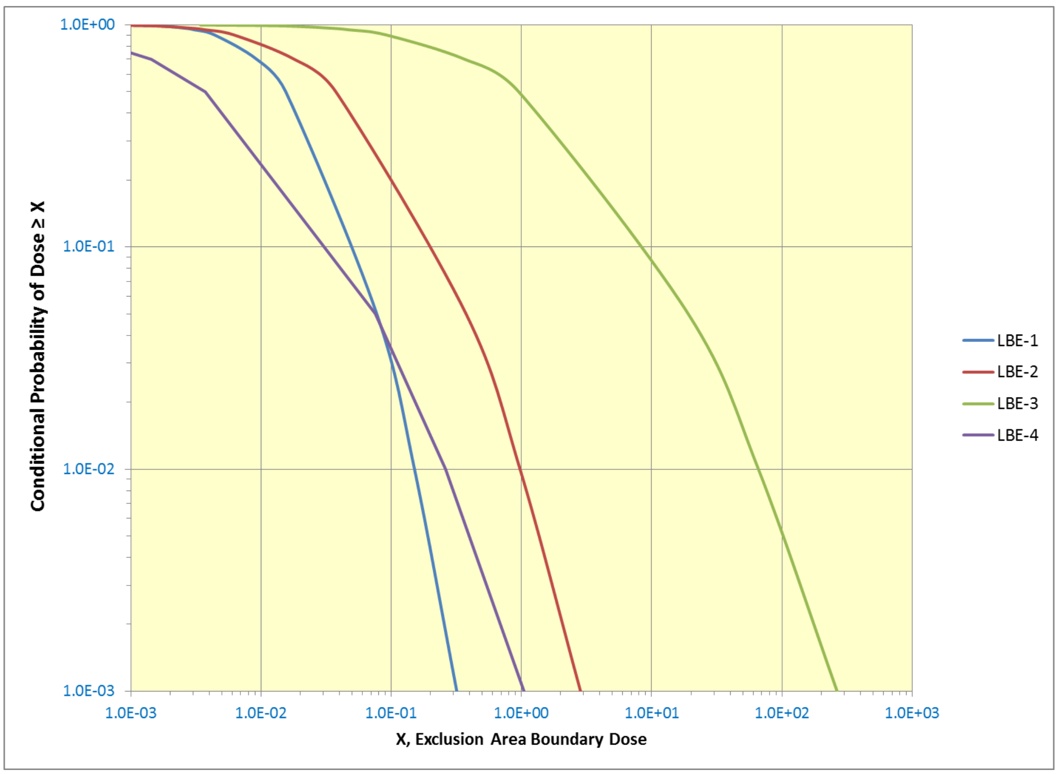
Where:

$F(d \geq x)$ is the total frequency of experiencing a dose greater than or equal to x

$P_j(d \geq x)$ is the conditional probability of experiencing a dose greater than or equal to x given occurrence of an event in event sequence family j. This is the complement of the cumulative uncertainty distribution for the dose estimate. The cumulative distribution is the probability that the dose is less than x, and its complement is the probability that it is equal to or greater than x.

The conditional probability of exceeding dose for the three example event sequence families is shown in Figure RI-1. These curves start at a probability of 1.0 because the probabilities are conditional on the occurrence of the events.

Fig. RI-1 $P_j(d \geq x)$, Conditional Probability of Exceeding Dose for Example Event Sequence Families



After weighing in the event sequence family frequencies, the exceedance frequency curve of Equation (2) can be developed and the results are shown in Figure RI-2. In constructing this figure the lognormal distributions for the conditional dose exceedance curves are truncated at 0.001. If a 1% threshold for relative risk significance is applied to the total exceedance frequency curve, it is seen from the figure that the relative risk significance of the event sequence families changes as the level of dose changes. For doses less than about 0.002 rem, only LBE-1 meets the 1% threshold. For doses between 0.002 and 0.15 rem, both LBE-1 and LBE-2 make risk-significant contributions. For doses between 0.15 rem and 0.32 rem, LBE-1, LBE-2, and LBE-3 are relative risk-significant. For doses between 0.32 rem and 2.84 rem, LBE-2 and LBE-3 are relative risk-significant, and for doses above 2.84 rem, only LBE-3 is relative risk-significant. LBE-4 does not meet the relative risk-significant threshold for any level of dose. These relative risk significance results are shown in Table RI-3.

Fig. RI-2 Dose Exceedance Frequency for Example PRA Results

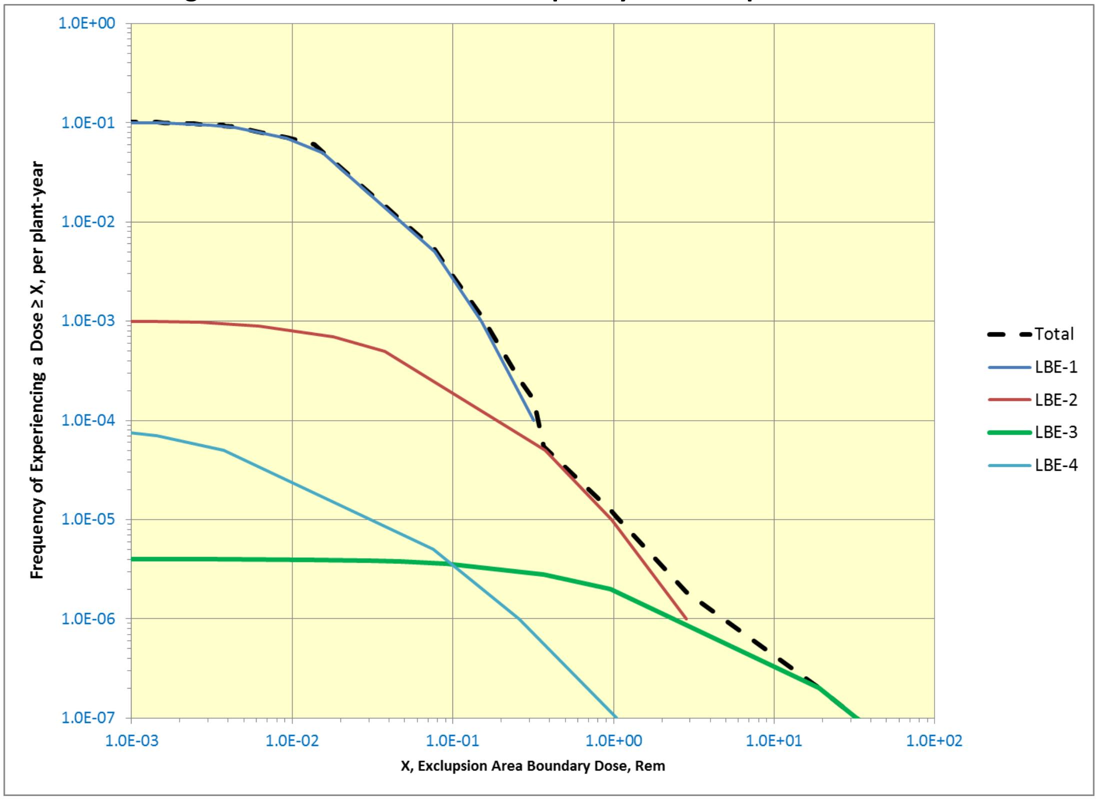


Table RI-3 Relative Risk Significance of Event Sequence Analysis Families Based on Dose Exceedance Frequency

EAB Dose Interval		Event Sequence Analysis Family Relative Risk-Significant?			
Lower	Upper	LBE-1	LBE-2	LBE-3	LBE-4
0.00001	0.002	Yes	No	No	No
0.002	0.15	Yes	Yes	No	No
0.15	0.32	Yes	Yes	Yes	No
0.32	2.84	No	Yes	Yes	No
2.84	265	No	No	Yes	No

RI.2.3 Conclusion

Two established methods for characterizing the risk have been compared for PRAs that provide quantitative estimates for both event sequence frequencies and consequences. Relative risk significance thresholds can be applied to both methods. When the point estimate is used to address the relative risk significance criteria for CC-I in SR RI-A2, LBE-1 and LBE-2 are classified as relative risk-significant. However, when using the exceedance frequency method as specified for CC-II in SR RI-A2, LBE-3 is also relative risk-significant and actually dominates the risk for large doses. LBE-4 is not relative risk-significant using both methods because it contributes less than 1% using both the point estimate method and the exceedance frequency method of expressing the total integrated risk.

The exceedance frequency method, which has been traditionally used to quantify the integrated risks from LWR Level 3 PRAs, explicitly accounts for the uncertainty in the consequence estimates, which include contributions from both mechanistic source term uncertainty and meteorological variability, enables the development of more refined risk insights compared with the mean point estimate method.

RI.3 AN EXAMPLE OF RISK INTEGRATION USING ABSOLUTE SIGNIFICANCE

The following sections provide examples of how absolute risk significance may be used to do the following:

(a) identify risk-significant event sequences and function performed by a system, structure, or component based on use of absolute risk significance criteria; and

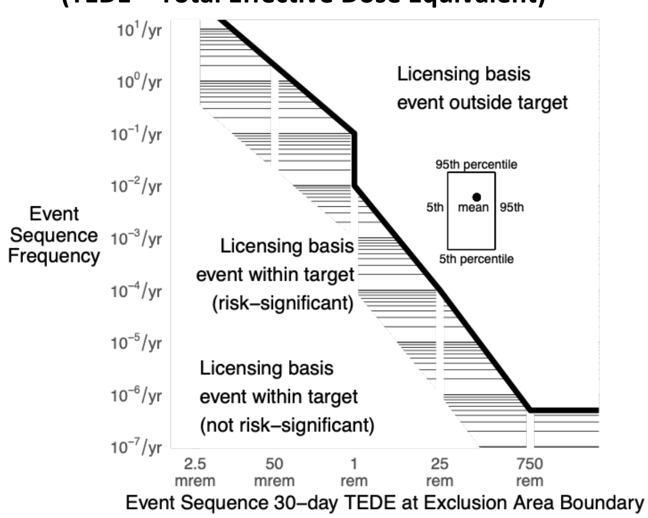
(b) conduct screening evaluations of event sequence families.

RI.3.1 Absolute Risk Significance Determination

The frequency-consequence target is a tool for determining the risk significance of basic events, functions, contributors, or SSCs. The risk significance of an SSC is actually the risk significance of an SSC in the performance of a specific function. When there are multiple basic events in the PRA model that represent causes of SSC failure or unavailability, the risk significance of the SSC is the risk significance of the union of the associated basic events. If the SSC performs multiple functions, the risk significance must be considered separately for each function. The risk significance of a function considers the possibility that there may be two or more SSCs that perform the same function. If the failure to perform a function is assumed, that includes all SSCs that perform that function are assumed to be failed. However, if the failure of an SSC to perform a specific function is assumed, only one SSC is involved in the evaluation of the risk significance of a specific function.

An event sequence family is considered within the frequency-consequence target when a point defined by both the event sequence family frequency and dose is within the frequency-consequence target. An example frequency-consequence target is presented in NEI 18-04 [RI-14], which defines the licensing basis events (i.e., licensing versions of event sequence families) with 95th percentile frequency and 95th percentile consequence values within one percent of the frequency-consequence target as being risk-significant as shown in Figure RI-3.

Fig. RI-3 Frequency-Consequence Target Curve from NEI 18-04 (TEDE = Total Effective Dose Equivalent)



The risk significance of a function performed by a SSC is depicted graphically by plotting on the frequency-consequence chart the event sequence family where a given system is successful. Next, an additional event sequence family is represented on the chart, this event sequence family being similar but where the system fails or is otherwise unavailable to perform its function.

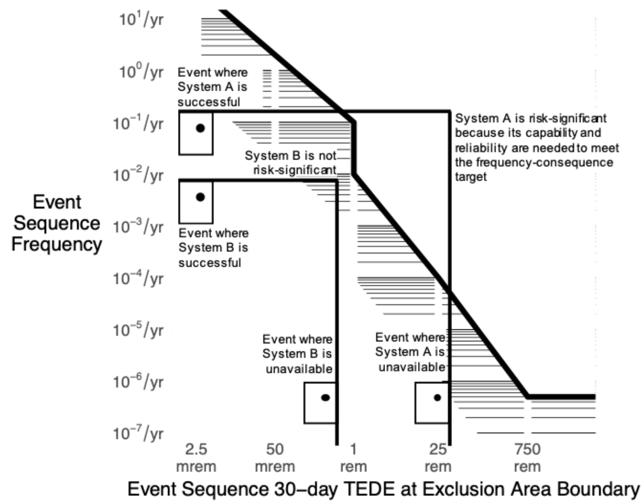
To assess the risk significance of the system, the hybrid point on the frequency-consequence chart is marked where the upper bound frequency of the event with system success is matched with the upper bound consequence of the event where the system is unavailable to simulate the impact of assuming failure of the system. When the function is assumed to be failed, the two LBEs, one involving success of the function and the other involving failure of the function, collapse to the hybrid point. If that point is beyond the frequency-consequence target, then that system was significant in keeping the event sequence family group within the target and is therefore risk-significant according to the absolute criteria in Table 1.9-2.

If the hybrid point connecting system success frequency to system failure consequence falls beneath the frequency-consequence target, then that system is not risk-significant according to this criterion. There are, however, additional criteria described below that could cause the system to be risk-significant even if the hybrid point falls below the frequency-consequence target in this exercise based upon the cumulative risk of all event sequence families. Figure RI-4 provides two examples to illustrate this concept:

(a) failure of the system would lead to that system being categorized as risk-significant because the event sequence hybrid point moves above the frequency-consequence target;

(b) failure of the system would not lead to that system being categorized as risk-significant because the event sequence hybrid point stays below the frequency-consequence target.

Fig. RI-4 Example Determination of Risk Significance against Frequency-Consequence Curve



The additional source of risk significance criteria is in the cumulative risk metrics for the plant. The SSC makes a significant contribution to one of the cumulative risk metrics used for evaluating the risk significance of event sequence families. A significant contribution to each cumulative risk metric limit is satisfied when the total frequency of all event sequence families with failure of the SSC exceeds 1% of the cumulative risk metric limit. This SSC risk significance criterion may be satisfied by an SSC whether or not it performs functions necessary to keep one or more event sequence families within the F-C Target. The cumulative risk metrics and limits include the following:

The total frequency of exceeding a site boundary dose of 100 mrem should not exceed 1/plant-year to ensure that the annual exposure limits in 10 CFR 20 are not exceeded. An SSC makes a significant contribution to this cumulative risk metric if the total frequency of exceeding a site boundary dose of 100 mrem associated with event sequence families with the SSC failed is greater than 10^{-2}/plant-year.

The average individual risk of early fatality within 1 mile of the EAB shall not exceed 5×10^{-7} per plant-year to ensure that the NRC safety goal Quantitative Health Objective (QHO) for early fatality risk is met. An SSC makes a significant contribution to this cumulative metric if the individual risk of early fatalities associated with the event sequence families with the SSC failed is greater than 5×10^{-9} per plant-year.

The average individual risk of latent cancer fatalities within 10 miles of the EAB shall not exceed 2×10^{-6} per plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met. An SSC makes a significant contribution to this cumulative risk metric if the individual risk of latent cancer fatalities associated with the event sequence families with the SSC failed is greater than 2×10^{-8} per plant-year.

The cumulative risk limit criteria in this SSC classification process are provided to address the situation in which an SSC may contribute to two or more event sequence families that collectively may be risk-significant even though the individual even sequences may not be significant. All event sequence families within the scope of the supporting PRA should be included when evaluating these cumulative risk limits. In such cases, the reliability and availability of such SSCs may need to be controlled to manage the total integrated risks over all the event sequence families.

RI.3.2 Absolute Risk Significance Event Sequence Family Screening

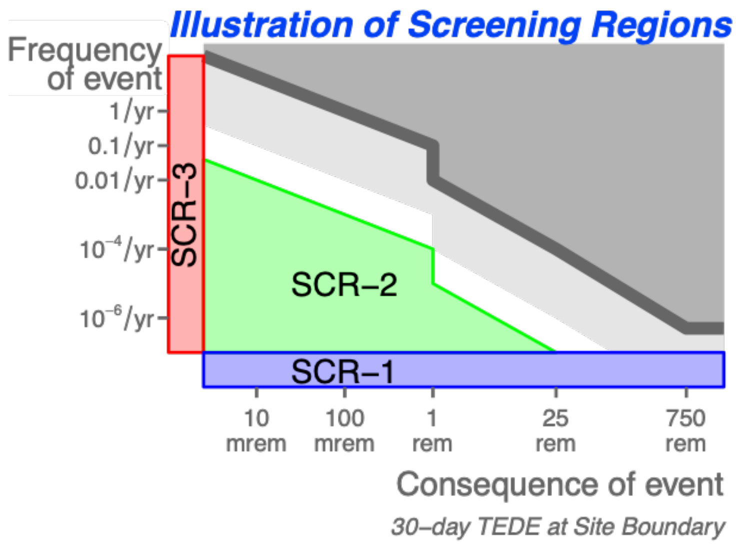
The screening criteria proposed in Table 1.10-1 allows for absolute, relative, and conservative screening based upon the risk metrics and criteria selected. Figure RI-5 shows how SCR-1, SCR-2, and SCR-3 would map to the NEI 18-04 F-C target curve [RI-14]. For narrative convenience, they are discussed in the order of SCR-1, SCR-3, and SCR-2:

(a) SCR-1: Absolute frequency screening—this screening criteria sets a lower limit on event sequence family frequencies of 10^{-7}/rx-yr. Event sequence families that can be shown to be less than this frequency can be screened from further consideration but should be retained in the model both for completeness checks and to ensure that future changes to the PRA do not increase this frequency above the screening threshold.

(b) SCR-3: Demonstrably conservative screening—this screening criteria is usually, but not always, conducted on the consequence axis of the F-C target. Event sequence families that can be shown to map to a release category before the start of the F-C target do not have a frequency target with which to determine risk significance. Thus, any event sequences with frequencies less than 2.5 mrem are screened from further analysis but should be retained in the model both for completeness checks and to ensure that future changes to the PRA do not increase this consequence above the screening threshold.

(c) SCR-2: Relative screening—this screening criteria is a hybrid of those set forth of SCR-1 and SCR-3. The concept behind SCR-2 is that the CC-I and CC-II modeling effort should be focused on the regions close (i.e., frequencies within three orders of magnitude) to the F-C target curve. When applying SCR-2, the analyst would either perform a demonstrably conservative consequence analysis or conservatively map the event sequence family to pre-calculated release categories. The analyst can then use this conservative consequence measure to determine the target frequency and from that frequency the screening frequency. If the event sequence family frequency can be shown to be less than the screening frequency, they can be screened from further consideration but should be retained in the model both for completeness checks and to ensure that future changes to the PRA do not increase this frequency above the screening threshold. These screened event sequence families are retained to support licensing basis event evaluations.

Fig. RI-5 Mapping of Three Screening Criteria to the NEI 18-04 F-C Target Curve



SECTION 5 - PRA CONFIGURATION CONTROL
CONTENTS

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SECTION 5
PRA CONFIGURATION CONTROL

5.1 PURPOSE

Section 5 states the requirements for a Configuration Control (CC) Program to support the use of a PRA in risk-informed decisions for nuclear power plants. Since these are administrative requirements, there is no gradation across Capability Categories. A discussion of the requirements is presented in this Section.

5.2 OBJECTIVE

The objectives of the Configuration Control Program are to ensure that when a PRA is to be used in risk-informed decisions, it represents the as-designed or as-built, as-operated or as-designed, as-intended-to-operate plant at the time of the decision. Furthermore, they ensure that any updates of the PRA are consistent with the technical requirements of this Standard.

The objectives of the Configuration Control Program ensure that

(a) a process for monitoring changes to the plant design, operation PRA technology, and industry experience are included;

(b) a process that maintains and upgrades the PRA to be consistent with the as-built, as-operated, or as-designed, as-intended-to-operate plant is included;

(c) the cumulative impact of pending changes in the performance of risk applications is considered;

(d) a process that maintains configuration control of computer codes and associated files used to support the PRA is included;

(e) the Configuration Control Program and its implementation are documented to provide traceability of the work.

For PRAs performed during a pre-operational stage, the user has the option to decide when the PRA has matured sufficiently to apply the requirements in this Section. These requirements shall be invoked prior to the performance of the first peer review according to the requirements in Section 6.

Table 5.8-1 High Level Requirements for Configuration Control

Designator	Requirement
HLR-CC-A	The PRA Configuration Control Program shall include a process for monitoring changes to the plant design, operation, PRA technology, and industry experience, and for collecting updated performance information that could result in changes to PRA inputs.
HLR-CC-B	The PRA Configuration Control Program shall include a process that maintains and upgrades the PRA to be consistent with the as-built, as-operated or as-designed, as-intended-to-operate plant.
HLR-CC-C	The PRA Configuration Control Program shall consider the cumulative impact of pending changes in the performance of risk applications.
HLR-CC-D	The PRA Configuration Control Program shall include a process that maintains configuration control of computer codes and associated files used to support PRA quantification.
HLR-CC-E	The documentation of the PRA Configuration Control Program and its implementation shall provide traceability of the work.

Table 5.8-2 Supporting Requirements for HLR-CC-A

The PRA Configuration Control Program shall include a process for monitoring changes to the plant design, operation, PRA technology, and industry experience, and for collecting updated performance information that could result in changes to PRA inputs. (HLR-CC-A)

Index No. CC-A	Capability Category I	Capability Category II
CC-A1	IMPLEMENT a process to track changes to the managed design configuration or to the physical plant, PRA technology, and related industry equipment performance/operational experience focused on collecting the necessary information to update PRA inputs.	
CC-A2	For operating plants, INCLUDE in the information collected the plant-specific changes in design, operation, and maintenance of the plant that impact, for example, the following: *(a)* operating procedures and practices (e.g., operations orders); *(b)* emergency and abnormal operating procedures; *(c)* design configuration; *(d)* initiating event frequencies; *(e)* system or subsystem unavailabilities; *(f)* component failure rates; *(g)* maintenance policies; *(h)* operator training; *(i)* technical specifications; *(j)* engineering calculations; *(k)* emergency plan; and *(l)* event management programs. See Note CC-N-1	
CC-A3	For operating plants, INCLUDE in the information collected changes to external facilities, sources of external hazards, or internal or external features that impact how external hazards may affect the plant. Such information may include, but is not limited to, the following: *(a)* changes in dam operating procedures that impact water release strategies; *(b)* regional changes that impact riverine flooding hazard analysis; and *(c)* capabilities of external response centers if such centers are credited in the PRA. See Note CC-N-1	
CC-A4	In the information collected, INCLUDE changes in industry experience that could impact *(a)* estimation of initiating event frequencies; *(b)* generic system or subsystem unavailabilities; *(c)* generic component failure rates; and *(d)* initiating events and their frequencies.	
CC-A5	In the information collected, INCLUDE changes to the PRA technology that could change the results of the PRA model.	
CC-A6	For PRAs performed during the pre-operational stage, INCLUDE in the collected information assumptions for the design, concept of operation, and concept of maintenance. Where possible, validate the assumptions using design information managed by a configuration management program. See Note CC-N-2	

Table 5.8-3 Supporting Requirements for HLR-CC-B

The PRA Configuration Control Program shall include a process that maintains and upgrades the PRA to be consistent with the as-built, as-operated or as-designed, as-intended-to-operate plant. (HLR-CC-B)

Index No. CC-B	Capability Category I	Capability Category II
CC-B1	EVALUATE changes in PRA inputs or of new information identified pursuant to the SRs of HLR-CC-A to determine whether such information warrants PRA maintenance or PRA upgrade. INCLUDE in the PRA changes identified per the SRs of HLR-CC-A.	
CC-B2	INCLUDE in the PRA those maintenance or upgrade changes implemented per the SRs of HLR-CC-A that would impact risk-significant insights.	
CC-B3	PERFORM a peer review of portions of the PRA that are affected by a PRA upgrade in accordance with the requirements specified in the Peer Review Section of each respective technical area of this Standard.	
CC-B4	ENSURE that changes to the PRA due to a PRA upgrade meet the requirements of the Technical Requirements Section of each respective technical area of this Standard.	
CC-B5	REVIEW changes made to the PRA affected by a PRA maintenance or upgrade in accordance with a utility-approved process.	

Table 5.8-4 Supporting Requirements for HLR-CC-C

The PRA Configuration Control Program shall consider the cumulative impact of pending changes in the performance of risk applications. (HLR-CC-C)

Index No. CC-C	Capability Category I	Capability Category II
CC-C1	DOCUMENT plant changes that have been identified to have a potential impact on PRA.	
CC-C2	DOCUMENT known industry issues or events and PRA technology changes that may have an impact on the PRA model.	

Table 5.8-5 Supporting Requirements for HLR-CC-D

The PRA Configuration Control Program shall include a process that maintains configuration control of computer codes and associated files used to support PRA quantification. (HLR-CC-D)

Index No. CC-D	Capability Category I	Capability Category II
CC-D1	ENSURE that the computer codes and associated files used to support and to perform PRA analyses are controlled to ensure consistent, reproducible results.	

Table 5.8-6 Supporting Requirements for HLR-CC-E

The documentation of the PRA Configuration Control Program and its implementation shall provide traceability of the work. (HLR-CC-E)

Index No. CC-E	Capability Category I	Capability Category II
CC-E1	DOCUMENT the process used in the Configuration Control Program, specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* a description of the process used to monitor PRA inputs and collect new information; *(b)* evidence that the aforementioned process is active; *(c)* descriptions of proposed and implemented changes; *(d)* a description of changes in a PRA due to each PRA upgrade or PRA maintenance; *(e)* a record of the performance and results of the appropriate PRA reviews (consistent with the requirements of Section 4.2); *(f)* a record of the process and results used to address the cumulative impact of pending changes; and *(g)* a description of the process used to maintain software configuration control.	
CC-E2	DOCUMENT the bases for the changes made to the PRA model.	

NONMANDATORY APPENDIX CC: NOTES AND EXPLANATORY MATERIAL FOR CONFIGURATION CONTROL

CC.1 NOTES ASSOCIATED WITH CONFIGURATION CONTROL

Table CC-1 Notes Supporting Configuration Control Requirements

Number	Notes
CC-N-1	This Supporting Requirement (SR) is likely not applicable to PRAs performed during the pre-operational stage. See CC-A2, CC-A3
CC-N-2	This SR is not applicable to operating plants. See CC-A6

SECTION 6 – PEER REVIEW
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SECTION 6
PEER REVIEW

6.1 PURPOSE

This Section states requirements for peer review of a PRA to be used in risk-informed decisions for advanced non-light water reactor nuclear power plants. Those portions of PRAs used for PRA applications applying this Standard shall be peer reviewed. The peer review shall assess the PRA to the extent necessary to determine if the method and its implementation meet the requirements of this Standard. Another purpose of the peer review is to determine the potential gaps in the PRA relative to this Standard's requirements. The peer review need not assess all aspects of the PRA against all the technical requirements in Section 4 of this Standard but must address all Supporting Requirements (SRs) relevant to the scope of the peer review. However, enough aspects of the PRA shall be reviewed for the reviewers to achieve consensus on the adequacy of the assessment of each applicable SR, as well as the methodologies and their implementation for each PRA technical element.

For PRAs performed during the pre-operational stage, the user has the option to decide when the PRA has matured sufficiently to apply the requirements in this Section.

For peer reviews on PRAs performed for plants in a design stage, or before plant operation, the peer review shall also address the assumptions that were made in lieu of as-built and as-operated design details, the judgments that were made as to which SRs could not be met because of the lack of such details, and whether the limitations of the PRA were adequately documented.

6.1.1 Documentation and Self-Assessment

A prerequisite for performing the peer review is that the PRA has been documented in analyses/calculations in the PRA that have been independently reviewed, and a self-assessment of the PRA has been conducted to establish the extent to which the PRA meets the requirements of this Standard. The results of the self-assessment process shall be documented.

6.1.2 Scope

Peer reviews shall be performed against the requirements in this Standard that are applicable to the plant operating state, sources of radioactive material, and hazard groups within the scope of the PRA that are being used to support risk-informed decisions. It is permissible to conduct a separate and distinct peer review for each relevant plant operating state, source of radioactive material, and hazard, or a single review that spans multiple or all plant operating states, sources of radioactive material, and hazards in the scope of the PRA. This Standard does not require that a single peer review be integrated across all hazard groups of the PRA and the relevant requirements of this Standard; however, an integration review shall address the question of whether the integration and aggregation across hazard groups were performed in accordance with the requirements in this Standard.

The scope of the peer review may be a "focused-scope" peer review. A focused-scope peer review is a subset of a complete full-scope peer review and involves specified technical elements and their associated High Level Requirements (HLRs) and SRs.

A focused-scope peer reviewed may be requested to

(a) support a specific application that does not involve the complete PRA model specific to identified plant operating states, radionuclide sources, and hazards;

(b) address changes to the PRA model as a result of upgrades; or

(c) serve to "close-out" findings from previous peer reviews that have been dispositioned.

When included in the scope of a peer review, a newly developed method shall be reviewed following the dedicated requirements discussed in Section 7.

The peer review is performed prior to using the PRA in risk-informed decisions. In addition, Section 5 of this Standard requires peer review for upgrades of a PRA. The scope of an additional peer review may be confined to changes to the PRA that have occurred since the previous review.

6.1.3 Process

The review shall be performed using a written method that assesses the PRA against the technical requirements in Section 4 of this Standard and addresses the requirements for each PRA element identified in Section 4 for each PRA technical element.

The peer review method shall consist of the following elements:

(a) a process for selection of the peer review team;

(b) training in the peer review process, including training on the unique aspects of the reactor technology in question;

(c) an approach to be used by the peer review team for assessing if the PRA meets the SRs of the technical requirements in Section 4 of this Standard;

(d) management and resolution of potential differing professional opinions;

(e) documentation of the results of the review.

When included in the scope of a peer review, a newly developed method shall be reviewed following the dedicated requirements discussed in Section 7.

6.2 PEER REVIEW TEAM COMPOSITION AND PERSONNEL QUALIFICATIONS

6.2.1 Collective Team

The peer review team shall consist of personnel whose collective qualifications include the following:

(a) the ability to assess all the PRA elements and subelements within the scope of the PRA, and the interfaces between those elements and subelements;

(b) the knowledge of the specific reactor technology; plant and reactor design; containment, confinement, or reactor building design; and plant operation.

6.2.2 Individual Team Members

(a) The peer review team members individually shall be

(1) knowledgeable of the requirements in this Standard for their area of review and how these requirements apply to the specific reactor technology;

(2) experienced in performing the activities related to the PRA elements for which the reviewer is assigned;

(3) independent from the team that developed the PRA model or the method being peer reviewed;

(4) subject matter experts that are included to judge the technical adequacy of non-PRA engineering evaluations and to confirm that the applicable envelope defining the limits of the method are identified

(5) prohibited from reviewing work performed by a direct supervisor or work they have directly supervised.

(b) If a peer review team member has performed or directly supervised work on any part of the PRA, that member shall not participate in the peer review of that part of the PRA or other parts that may have been influenced by the member's work. In addition, the peer reviewer shall not be assigned an area for review where their current immediate supervisor performed the actual technical analysis.

(c) A peer reviewer shall not be assigned a newly developed method to review if the reviewer was an author or co-author of the method under consideration, or their current immediate supervisor was an author or co-author of the method under consideration.

6.2.3 Specific Review Team Qualifications

The peer reviewer shall also be knowledgeable (by direct experience) of the specific method, code, tool, or approach (e.g., large event tree linking approach, specific computer codes that were used, human reliability analysis method, etc.) that was used in the PRA element assigned for review. Understanding and competence in the assigned area shall be demonstrated by the range of the individual's experience in the number of different, independent activities performed in the assigned area, as well as the different levels of complexity of these activities.

(a) One member of the peer review team (the technical integrator) shall be familiar with all the PRA elements identified in the relevant section of this Standard under review and shall have demonstrated the capability to integrate these PRA elements. When more than one portion of the PRA (e.g., hazard group, radionuclide source, or plant operating state) is under review, a separate technical integrator may be used for each portion.

(b) The peer review team shall have a team leader to lead the team in the performance of the review. The team leader need not be the technical integrator.

(c) The peer review should be conducted by a team with a minimum of five members performed over a minimum period of one week, unless a shorter review is justified in the peer review documentation. If the scope of the PRA to be reviewed includes mechanistic source terms and/or radiological consequence analyses, larger review teams may be required. Smaller review teams may be used for limited-scope PRAs performed during a design stage prior to regulatory submittal. If the review is focused on a particular PRA element, such as a review of an upgrade of a PRA element, then the peer review should be conducted by a team with a minimum of two members, performed over a time necessary to address the specific PRA element.

(d) Exceptions to the requirements of this paragraph may be taken based on the availability of appropriate personnel to develop a team. A single-person peer review shall only be justified when the review involves an upgrade of a single element and the reviewer has acceptable qualifications for the technologies involved in the upgrade. All such exceptions shall be documented. Regardless of any such exceptions, the collective qualification of the review team shall be appropriate to the full scope of SRs within the scope of the hazard group PRA being peer reviewed.

(e) If the peer reviewer is reviewing a newly developed method, the reviewer shall be knowledgeable of the technical area addressed by the newly developed method. Understanding and competence of the newly developed method shall be demonstrated by the range of the individual's experience in that technical area, that is, the years and number of different activities performed in the technical area, as well as the different levels of complexity of the technical area. Subject matter experts should be included to judge the technical adequacy of non-PRA engineering evaluations and to confirm that the applicable envelope defining the limits of the method are identified.

6.3 REVIEW OF PRA ELEMENTS TO CONFIRM THE METHODOLOGY USED AND IMPLEMENTATION

The peer review team shall use the requirements outlined in this Section, as complemented by the element-specific review requirements in the respective section for each of the PRA elements within the scope of the PRA being reviewed to determine if the methodology and the implementation of the methodology for each PRA element meet the requirements of this Standard. Additional material for those elements may be reviewed depending on the results obtained. These suggestions are not intended to be a minimum or comprehensive list of requirements. The judgment of the reviewer shall be used to determine the specific scope and depth of the review in each PRA element.

For PRAs performed during the pre-operational stage or PRAs performed on a bounding site, and for each PRA element, the reviewer(s) shall determine if the level of detail of the PRA model is consistent with the level of detail of the design information available to support, and referenced by, the PRA. The PRA documentation shall be reviewed to confirm that any assumptions made or limitations of the PRA model due to the lack of design information are sufficiently documented.

The results of the PRA, including the models, assumptions, and results of each PRA element within the scope of the review, shall be reviewed to determine their reasonableness given the design and operation of the plant (e.g., investigation of cutset or sequence combinations for reasonableness). The limitations of the PRA due to lack of design information shall also be within the scope of the review.

Any newly developed method included in the scope of the peer review is reviewed against the requirements of Section 7. It is noted that a newly developed method can be peer reviewed within the scope of a plant PRA (i.e., concurrently with its implementation in a plant PRA) or via a dedicated stand-alone peer review. If newly developed methods are peer reviewed concurrently with the implementations of methods, all specific requirements for the newly developed methods peer review shall be met. In this second case, the implementation of the method is peer reviewed in a separate peer review.

Even if exceptions to the review team qualification requirements occur, concerning the composition of the peer review team or the duration of the review, all SRs relevant to the scope of the peer review of the PRA are to be reviewed.

The extent of a focused-scope peer review includes all SRs (e.g., not just those for which findings were cited), within the HLR(s) containing SRs with findings. New findings may be issued even for SRs that did not have previous findings since a focused-scope peer review encompasses all the SRs within an affected HLR.

6.4 EXPERT JUDGMENT

The use of expert judgment to implement requirements in this Standard shall be reviewed using the general requirements in Section 4.2.

6.5 PRA CONFIGURATION CONTROL

The peer review team shall review the process, including implementation, for maintaining or upgrading the PRA against the Configuration Control requirements of this Standard in Section 5.

6.6 PEER REVIEW DOCUMENTATION

6.6.1 Peer Review Team Documentation

The peer review team's documentation shall demonstrate that the review process appropriately implemented the review requirements.

Specifically, the peer review documentation shall include the following:

(a) identification of the version of the PRA reviewed;

(b) a statement of the scope of the peer review;

(c) the names of the peer review team members;

(d) a brief resume on each team member describing the individual's employer, education, PRA training, and PRA element experience and expertise;

(e) the elements of the PRA reviewed by each team member;

(f) a discussion of the extent to which each PRA element was reviewed including justification for any SRs within the scope that were not reviewed;

(g) results of the review identifying any differences between the requirements in the technical and Configuration Control requirements in Sections 4 and 5 of this Standard and the methodologies implemented, defined to a sufficient level of detail that will allow the resolution of the differences;

(h) identification and significance of exceptions and gaps relative to this Standard's requirements in sufficient detail to allow the resolution of the gaps that the peer reviewers have determined to be important to the PRA;

(i) an assessment of the PRA assumptions that the reviewers have determined to be important to the PRA;

(j) at the request of any peer reviewer, differences or dissenting views among peer reviewers;

(k) recommended alternatives for resolution of any differences;

(l) an assessment of the Capability Category of the SRs (i.e., identification of what Capability Category is met for the SRs);

(m) identification of any new PRA method reviewed, including the criteria (bases) used to determine acceptability of method;

(n) results of the review of a newly developed method, consistent with Section 7;

(o) an assessment of the adequacy of documentation of limitations of the PRA and justification for why certain HLRs or SRs could not be addressed due to lack of design and operational information for PRAs performed on plants prior to design completion, construction, or operation.

6.6.2 Resolution of Peer Review Team Comments

Resolution of deficiencies against the requirements of this standard that are identified by the Peer Review Team shall be documented. The resolution of these deficiencies shall describe how each was addressed such that the associated SR can now be demonstrated to be met. The documentation shall indicate whether the deficiency is resolved via a PRA maintenance or a PRA upgrade. The determination of whether the resolution adequately eliminates the deficiency shall be made by one or more individuals who meet the qualification requirements of Section 6.2.2.

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SECTION 7 – NEWLY DEVELOPED METHODS
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SECTION 7
NEWLY DEVELOPED METHODS

7.1 INTRODUCTION

This Section states requirements for Newly Developed Methods (NM) explicitly developed for use in PRA to support risk-informed decisions for nuclear power plants. The High Level Requirements (HLRs) and Supporting Requirements (SRs) for the Newly Developed Methods are contained in this Section. For PRAs performed on plants in the pre-operational stage, these requirements apply for newly developed methods that are introduced following the first peer review performed according to the requirements in Section 6.

7.2 OBJECTIVES AND TECHNICAL REQUIREMENTS FOR NEWLY DEVELOPED METHODS (NM)

The objectives of the Newly Developed Methods requirements are to ensure that a newly developed method is technically adequate and are as follows:

(a) the Newly Developed Methods have clearly defined scope and limitations;

(b) the Newly Developed Methods are based on sound engineering and relevant science;

(c) the Newly Developed Methods have proper treatment of assumptions and uncertainties;

(d) the Newly Developed Methods are based on appropriate and well understood data;

(e) the Newly Developed Methods produce results that are consistent with expectations;

(f) the Newly Developed Methods are clearly documented in such a way that knowledgeable personnel can understand them without ambiguity and that there is enough documentation so that it can be peer reviewed.

The objectives above are intended to be applicable to a large spectrum of methods, although it is understood that not all the SRs could be applicable to all methods. In some cases, depending on the method scope and purpose, some of the SRs may not be applicable. In addition, the SRs are designed to be able to address a stand-alone method (i.e., independent from its implementation on a specific plant PRA). It is recognized that, in some circumstances, a method can be so plant- or site-specific (especially in the external hazard domain) that a full review of the method can only be performed within its implementation. In such cases, it is envisioned that some of the Newly Developed Methods SRs could be overlapping with technical element specific SRs. In such cases, the SRs in the appropriate technical element may take priority to some Newly Developed Methods SRs.

Table 7.2-1 High Level Requirements for Newly Developed Methods

Designator	Requirement
HLR-NM-A	The purpose and scope of the Newly Developed Methods shall be clearly stated.
HLR-NM-B	The Newly Developed Methods shall be based on sound engineering and science relevant to its purpose and scope.
HLR-NM-C	The data (note that data can be numeric or non-numeric in nature) shall be relevant Newly Developed Methods, technically sound, and properly analyzed and applied.
HLR-NM-D	Uncertainties in the Newly Developed Methods shall be characterized. Sources of model uncertainties and related assumptions shall be identified.
HLR-NM-E	The results of the Newly Developed Methods shall be reproducible, reasonable, and consistent with the assumptions and data, and given the purpose and scope of the Newly Developed Methods.
HLR-NM-F	The documentation of the Newly Developed Methods shall provide traceability of the work and facilitate incorporation of the Newly Developed Methods in a PRA model.

Table 7.2-2 Supporting Requirements for HLR-NM-A

The purpose and scope of the Newly Developed Methods shall be clearly stated. (HLR-NM-A)

<table>
<tr><th>Index No. NM-A</th><th>Capability Category I</th><th>Capability Category II</th></tr>
<tr><td>NM-A1</td><td colspan="2">ENSURE that the stated purpose of the Newly Developed Methods (i.e., what is being achieved by the Newly Developed Methods) is consistent with the scope (established boundary) of the Newly Developed Methods.</td></tr>
<tr><td>NM-A2</td><td colspan="2">ENSURE that the applicability and limitations of the Newly Developed Methods are consistent with the purpose and scope in Requirement NM-A1.</td></tr>
<tr><td>NM-A3</td><td colspan="2">IDENTIFY the limitations and applicability of the Newly Developed Methods and the areas of the PRA for which the Newly Developed Methods is intended to be used, and those for which it is specifically not intended (e.g., hazards, technical elements, plant features, SRs impacted by the Newly Developed Methods).</td></tr>
</table>

Table 7.2-3 Supporting Requirements for HLR-NM-B

The Newly Developed Methods shall be based on sound engineering and science relevant to its purpose and scope. (HLR-NM-B)

<table>
<tr><th>Index No. NM-B</th><th>Capability Category I</th><th>Capability Category II</th></tr>
<tr><td>NM-B1</td><td colspan="2">ESTABLISH the technical bases for the Newly Developed Methods by using approaches founded on established mathematical, engineering, and/or scientific principles (e.g., established through operating experience, tests, benchmarking, or acceptance by the scientific community).</td></tr>
<tr><td>NM-B2</td><td colspan="2">If empirical models are used, ENSURE that they are supported by sufficient data which are relevant to the Newly Developed Methods and, to the extent possible, that the experimental data have been shown to be repeatable.</td></tr>
<tr><td>NM-B3</td><td colspan="2">IDENTIFY assumptions used to develop the technical bases of the Newly Developed Methods.</td></tr>
<tr><td>NM-B4</td><td colspan="2">JUSTIFY the rationale for the assumptions identified in Requirement NM-B3 (e.g., backed by appropriate operational experience).</td></tr>
</table>

Table 7.2-4 Supporting Requirements for HLR-NM-C

The data (note that data can be numeric or non-numeric in nature) shall be relevant to the Newly Developed Methods, technically sound, and properly analyzed and applied. (HLR-NM-C)

<table>
<tr><th>Index No. NM-C</th><th>Capability Category I</th><th>Capability Category II</th></tr>
<tr><td>NM-C1</td><td colspan="2">IDENTIFY the data needed in the development of the Newly Developed Methods (e.g., relevant plant-specific data, industry-wide current operating experience and data, or experimental or test data).</td></tr>
<tr><td>NM-C2</td><td colspan="2">COLLECT relevant data consistent with current technical state-of-practice.</td></tr>
<tr><td>NM-C3</td><td colspan="2">DEMONSTRATE that the data used, including experimental data or test data, is relevant to and supports the technical basis of the Newly Developed Methods.</td></tr>
<tr><td>NM-C4</td><td colspan="2">SPECIFY the basis for exclusion of data identified in Requirement NM-C1.</td></tr>
<tr><td>NM-C5</td><td colspan="2">ANALYZE data (e.g., modifications to the data, use of data in a different context or beyond the original ranges, statistical analysis) using technically sound basis or criteria.</td></tr>
<tr><td>NM-C6</td><td colspan="2">ENSURE that data is applied consistent with the purpose and scope of the Newly Developed Methods.</td></tr>
</table>

Table 7.2-5 Supporting Requirements for HLR-NM-D

Uncertainties in the Newly Developed Methods shall be characterized. Sources of model uncertainties and related assumptions shall be identified. (HLR-NM-D)

Index No. NM-D	Capability Category I	Capability Category II
NM-D1	CHARACTERIZE the parameter uncertainties associated with the Newly Developed Methods consistent with the intended scope and purpose of the method; this characterization could include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the parameter estimate as conservative or bounding. See Note NM-N-1	
NM-D2	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives associated with assumptions identified in Requirement NM-B3.	
NM-D3	CHARACTERIZE the model uncertainties (identified in Requirement NM-D2) associated with the Newly Developed Methods; this characterization could be in the form of sensitivity studies.	

Table 7.2-6 Supporting Requirements for HLR-NM-E

The results of the Newly Developed Methods shall be reproducible, reasonable, and consistent with the assumptions and data, and given the purpose and scope of the Newly Developed Methods. (HLR-NM-E)

Index No. NM-E	Capability Category I	Capability Category II
NM-E1	REVIEW the results from the Newly Developed Methods to determine that they are reproducible, reasonable, and consistent with assumptions and data addressed in the SRs under HLR-NM-B and HLR-NM-C.	
NM-E2	COMPARE the results of the Newly Developed Methods with existing methods and, when possible, IDENTIFY causes for substantial differences.	
NM-E3	ENSURE uncertainties do not preclude meaningful use of the Newly Developed Methods results.	

Table 7.2-7 Supporting Requirements for HLR-NM-F

The documentation of the Newly Developed Methods shall provide traceability of the work and facilitate incorporation of the Newly Developed Methods in a PRA model. (HLR-NM-F)

Index No. NM-F	Capability Category I	Capability Category II
NM-F1	DOCUMENT the process used in the Newly Developed Methods, specifying what is used as input, the technical basis, and the implementation limitations. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the purpose and scope of the Newly Developed Methods; *(b)* the intended use of the Newly Developed Methods; *(c)* the limitations of the Newly Developed Methods; *(d)* the technical basis for the Newly Developed Methods; *(e)* the sources of data, and the collection process in support of the Newly Developed Methods; *(f)* the assumptions and uncertainties associated with the Newly Developed Methods; and *(g)* the interpretation of the results of the Newly Developed Methods in the framework of the intended use and application.	
NM-F2	DOCUMENT the intended process by which the Newly Developed Methods can be applied to a PRA model consistently with the intended use of the Newly Developed Methods and taking into account the purpose, scope, and limitations.	

NONMANDATORY APPENDIX NM: NOTES AND EXPLANATORY MATERIAL FOR NEWLY DEVELOPED METHODS

NM.1 NOTES ASSOCIATED WITH NEWLY DEVELOPED METHODS

Table NM-1 Notes Supporting Newly Developed Methods Requirements

Number	Notes
NM-N-1	Depending on the purpose and scope of the method, uncertainty distributions may need to be explicitly calculated to allow for application of a method for risk-significant items to meet Capability Category II (CC-II) of related SRs in other technical elements of this Standard. See NM-D1

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