

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

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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.

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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

1.1 OBJECTIVE

This Standard¹ 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 listed³:

- (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).

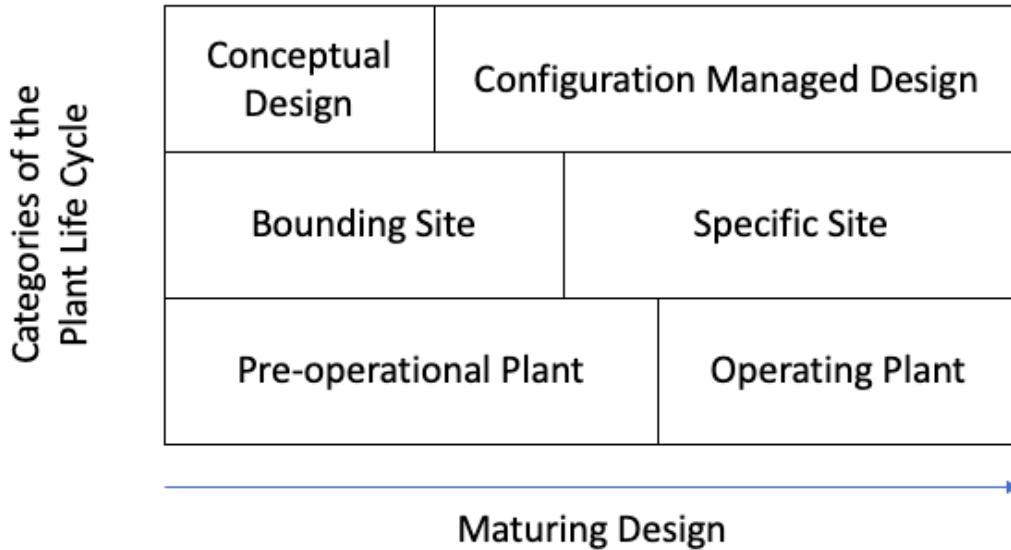
When a specific PRA application is undertaken, judgment is needed to determine which Capability Category is needed for each portion of the PRA, and hence, which SRs apply to the PRA applications.

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 1×10^{-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, MSR, 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

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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 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., $<1\text{E-6/plant-yr}$).

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 *hazards* 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., $<1\text{E-4/plant-yr}$).

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/s²).

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

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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).

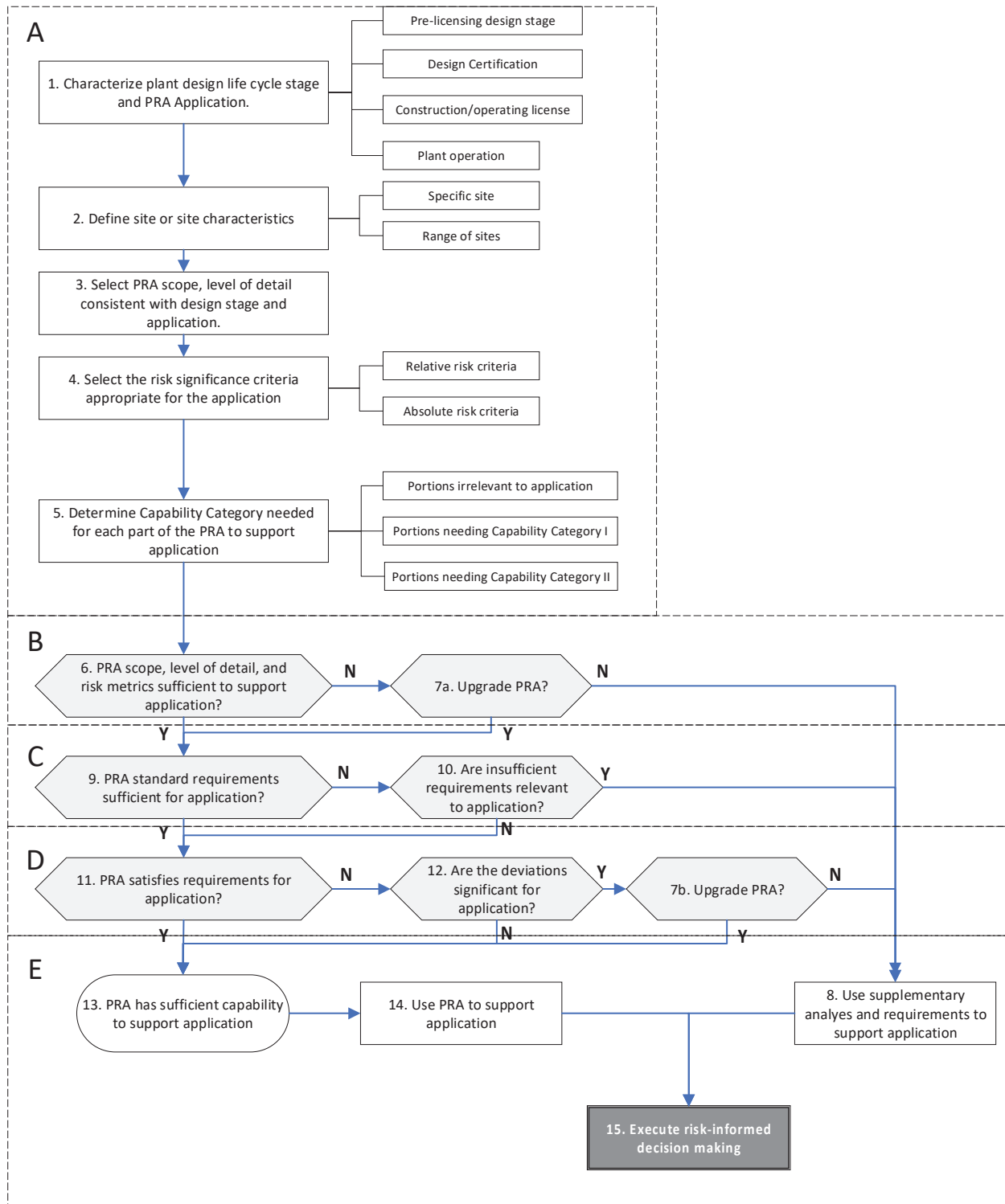
(c) Stage C. An evaluation is performed to determine whether the capability requirements for the SRs from this Standard for each relevant portion of the PRA are sufficient to support the application. If not, the SRs may be augmented with supplementary requirements as described in Stage E. For plants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