

ASME/ANS RA-S-1.2–2024

(Revision of Trial Use Standard 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)

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CONTENTS

Foreword	v
Acknowledgments	vii
ASME/ANS RA-S Committee	ix
Correspondence with the ASME/ANS Joint Committee on Nuclear Risk Management	xi
PART 1	GENERAL REQUIREMENTS FOR A LEVEL 2 PRA. 1
Section 1-1	Introduction. 1
Section 1-2	Acronyms and Definitions 8
Section 1-3	PRA Scope and Capabilities in Support of Risk-Informed Applications 15
Section 1-4	Requirements for Use of Expert Judgment. 16
Section 1-5	PRA Configuration Control Program 17
Section 1-6	Peer Review. 20
Section 1-7	Newly Developed Methods 23
Section 1-8	References 26
Nonmandatory Appendix	27
1-A	Meanings of Action Verbs 27
PART 2	TECHNICAL REQUIREMENTS FOR LEVEL 2 PRA. 30
Section 2-1	Overview of Level 2 PRA Requirements. 30
Section 2-2	Level 2 PRA Technical Elements and Requirements 31
Section 2-3	References 59
Nonmandatory Appendix	60
2-A	Explanatory Notes Regarding Application and Review of The Level 2 Supporting Requirements 60

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FOREWORD

The American Society of Mechanical Engineers (ASME) Board on Nuclear Codes and Standards (BNCS) and American Nuclear Society (ANS) Standards Board have formed a Joint Committee on Nuclear Risk Management (JCNRM) to develop and maintain probabilistic risk assessment (PRA) standards. The JCNRM operates under procedures accredited by the American National Standards Institute (ANSI) as meeting the criteria of consensus procedures for American National Standards. The JCNRM holds two formal meetings per year, and users are invited to participate. Additional information about the JCNRM can be found on its committee page at <https://go.asme.org/JCNRMcommittee>.

This Level 2 Standard, ASME/ANS RA-S-1.2-2024, “Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs),” was initiated via Project Number ANS-58.24, which ANS later formally requested ANSI to transfer to ASME. In 2015, the JCNRM published 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),” which was the trial-use and pilot-application version of this Standard. After a two-year trial-use and pilot-application period, all comments received were collected and resolved by the Level 2 Working Group within the JCNRM. In addition to resolving comments received during the trial-use and pilot-application period, this Standard has been updated to reflect updates that have been made to ASME and JCNRM writers’ guides and ballot comments received from the JCNRM. These resulted in a number of changes being made to support self-consistency as well as consistency with other JCNRM standards.

This Standard, ASME/ANS RA-S-1.2-2024, is the current edition of the Level 2 PRA Standard that supersedes all previous versions. The JCNRM is responsible for ensuring that this Standard is maintained and revised, as necessary. This responsibility includes appropriate coordination with and linkage to other standards under development for related risk-informed applications.

ASME/ANS RA-S-1.2-2024 is a substantial revision of the trial-use and pilot-application Standard, ASME/ANS RA-S-1.2-2014. The following major modifications are among those performed:

- The Level 3 Interface Technical Element has been removed on the basis that the Level 3 trial-use and pilot-application Standard, ASME/ANS RA-S-1.3-2017, “Standard for Radiological Accident Offsite Consequence Analysis (Level 3 PRA) to Support Nuclear Installation Applications,” includes all necessary requirements to properly transfer information from a Level 2 PRA.
- Supporting Requirements (SRs) with “No Requirement” for Capability Category I have been redefined in such a way that it is now clear what the requirements are to meet each Capability Category.
- SRs that reference back to the Level 1 Standard, ASME/ANS RA-S-1.1-2024, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” have been made more consistent, deliberate, and explicit in each Part to facilitate the peer-review process.
- [Part 1](#) has been substantially revised in order to be consistent with the Level 1 Standard and includes revised definitions of significance, new sections dedicated to Configuration Control and Newly Developed Methods, and a [Nonmandatory Appendix \(NMA\) 1-A](#) that defines all action verbs used in this Standard.
- Capability Category III has been removed across the board on the basis that Capability Category II already envisions refined analysis and realism implemented for the risk-significant elements. Going beyond this, while not discouraged, is not something that needs to be codified in a standard that is supposed to identify the minimum requirements for a technically adequate analysis.
- A number of changes have been implemented to strengthen the consistency among technical elements that are cross-cutting through different standards developed by the JCNRM. These

changes required, for example, revisiting SRs associated with screening, uncertainty, human reliability analysis, and documentation.

- Notes and commentaries have been revised to ensure content is still up to date and, for the most part, are removed from the body of this Standard and located in the [NMA 2-A](#). This relocation emphasizes the concept that notes and commentaries do not represent formal requirements of this Standard and are provided for information. References are also removed from individual SRs and moved to notes as one way to meet the SRs.

This publication, ASME/ANS RA-S-1.2-2024, “Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs),” was approved by the ASME BNCS and the ANS Standards Board. ASME/ANS RA-S-1.2-2024 was approved by ANSI on May 31, 2024.

ACKNOWLEDGMENTS

The ANS/ASME JCNRM is animated by the passion of more than 200 professionals in the industry, from four continents and spanning the extensive interdisciplinary breadth needed for the development of multihazard, full-scope, comprehensive risk assessments. Their dedication and support continue to sustain the primary role that risk information has in the safe and efficient design, operation, and regulation of nuclear power plants. The members of the JCNRM Nuclear Risk Standards and Guidance Subcommittee and the JCNRM Technical Requirements Subcommittee, including reporting working groups, have dedicated significant time to the refinement of this Standard.

A particular debt of gratitude is owed by the JCNRM to Ray Schneider and N. Reed LaBarge, who have been instrumental in leading and coordinating the combined effort needed to update and edit this edition of the Standard, navigating the schedule and challenges of a volunteer organization while maintaining the highest technical rigor.

A number of people have supported the JCNRM for numerous years but retired before seeing the completion of this Standard, for which they provided instrumental help. We acknowledge the efforts of these people and especially the work of Ed Burns, former Level 2 Working Group chair.

We also remember dear friends and significant contributors to this Standard and to the risk-informed technology community that have passed. In memoriam, we acknowledge Mary Drouin, Barry Sloane, and Rupert Weston.

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Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs)

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Revisions and Errata. The committee processes revisions to this Standard on a continuous basis to incorporate changes that appear necessary or desirable as demonstrated by the experience gained from the application of the Standard. Approved revisions will be published in the next edition of the Standard.

In addition, the committee may post errata on the committee web page. Errata become effective on the date posted. Users can register on the committee web page to receive e-mail notifications of posted errata.

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Cases

(a) The most common applications for cases are

- (1) to permit early implementation of a revision based on an urgent need
- (2) to provide alternative requirements
- (3) to allow users to gain experience with alternative or potential additional requirements prior to incorporation directly into the Standard
- (4) to permit the use of a new material or process

(b) Users are cautioned that not all jurisdictions or owners automatically accept cases. Cases are not to be considered as approving, recommending, certifying, or endorsing any proprietary or specific design, or as limiting in any way the freedom of manufacturers, constructors, or owners to choose any method of design or any form of construction that conforms to the Standard.

(c) A proposed case shall be written as a question and reply in the same format as existing cases. The proposal shall also include the following information:

- (1) a statement of need and background information
- (2) the urgency of the case (e.g., the case concerns a project that is underway or imminent)
- (3) the Standard and the paragraph, figure, or table number
- (4) the editions of the Standard to which the proposed case applies

(d) A case is effective for use when the public review process has been completed and it is approved by the cognizant supervisory board. Approved cases are posted on the committee web page.

Interpretations. Upon request, the committee will issue an interpretation of any requirement of this Standard. An interpretation can be issued only in response to a request submitted through the online Interpretation Submittal Form at <https://go.asme.org/InterpretationRequest>. Upon submitting the form, the inquirer will receive an automatic e-mail confirming receipt.

ASME does not act as a consultant for specific engineering problems or for the general application or understanding of the Standard requirements. If, based on the information submitted, it is the opinion of the committee that the inquirer should seek assistance, the request will be returned with the recommendation that such assistance be obtained. Inquirers can track the status of their requests at <https://go.asme.org/Interpretations>.

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Committee Meetings. The ASME/ANS JCNRM regularly holds meetings that are open to the public. Persons wishing to attend any meeting should contact the secretary of the committee. Information on future committee meetings can be found on the committee web page at <https://go.asme.org/JCNRMcommittee>.

PART 1

GENERAL REQUIREMENTS

FOR A LEVEL 2 PRA

Section 1-1

Introduction

1-1.1 OBJECTIVE

This Standard states the requirements for Level 2 probabilistic risk assessments (PRAs) for severe accident progression and radiological release for use in supporting risk-informed decisions for commercial light water reactor (LWR) nuclear power plants.

1-1.2 SCOPE AND APPLICABILITY

The scope of this Standard is limited to analyzing the progression of severe accidents from the onset of core damage through radionuclide release to the environment or a determination that a release to the environment will not occur. It includes the analysis of the various phenomena that occur inside the reactor vessel, the containment structure, and neighboring structures that might participate in the radiological release pathway to the environment. This analysis involves carrying the postulated accident sequences from a Level 1 PRA through a probabilistic logic structure such as a containment event tree (CET) (or equivalent) and determining the radionuclide release characteristics (e.g., magnitude and timing) for the various pathways through the CET.

This scope includes accident sequences initiated by internal events, internal hazards, and/or external hazards addressed in ASME/ANS RA-S-1.1-2024 [1-1]. It also includes postulated accident sequences initiated from all modes of reactor operation (at-power,

shutdown, and transition states) addressed in ASME/ANS-58.22-2014 (Shutdown PRA Trial Use Standard) [1-2].

The assessment of radiological releases is restricted to radionuclides that originate in fuel located within the reactor pressure vessel. It does not address spent fuel pool radionuclide release nor releases related to purposeful human-induced security threats (e.g., sabotage); this limited scope is consistent with that of ASME/ANS RA-S-1.1-2024 [1-1]. This Standard is limited in scope to single reactor accidents and does not address accident sequences involving releases and interactions among multi-reactor units and fuel storage facilities such as those which occurred at Fukushima Daiichi during March 2011. However, the Standard does provide requirements regarding the status of any other units on site (e.g., status of shared systems inherited from the Level 1 PRA). As such, multi-unit issues are treated to a limited extent, broadly consistent with ASME/ANS RA-S-1.1-2024 [1-1], but combined accident sequences, multiple simultaneous releases from multiple units and radiological sources, and more complex multi-unit issues and interactions are not addressed within this Standard.

These requirements are written for operating LWR power plants (i.e., plants with designs and features similar to the plants operating when this Standard was published). They may be used for LWR plants under design or construction or for advanced LWRs, but revised or additional requirements may be needed.

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

1-1.2.1 Interface with Level 1 PRA Standards

Requirements of this Standard are applicable to all hazards developed following the requirements of ASME/ANS RA-S-1.1-2024 [1-1] and are intended to be compatible with plant damage states that may be characterized from results of PRA analyses generated in accordance with ASME/ANS-58.22-2014 (Trial Use) [1-2]. ASME/ANS RA-S-1.1-2024 [1-1] specifically includes PRA requirements for internal events, internal hazards and external hazards that might occur while the nuclear power plant is at-power. This Standard also may be used to develop the Level 2 PRA for use with plant operating states (POSs) defined in the ASME/ANS-58.22-2014 (Shutdown PRA Trial Use Standard) [1-2].

Use of this Standard does not require specific Level 1 scope be included (e.g., hazard models). However, hazard-specific features needed to develop human actions, system models and containment failure mechanism are identified within individual Supporting Requirements (SRs) that should be included when those hazards are within the scope of the Level 2 PRA.

1-1.2.2 Interface with Level 3 PRA Standard

The end point of a Level 2 analysis is the distribution of the plant damage states from the Level 1 PRA into a set of radionuclide release categories (RCs). These RCs and their characterization represent a critical input to the Level 3 PRA. This Standard, therefore, specifies the requirements for an analysis sufficient to characterize the RCs (i.e., frequency, magnitude, and timing of fission product releases) for use in a Level 3 PRA. Requirements for performing the Level 3 PRA are contained in ASME/ANS-RA-S-1.3-2017 [1-3].

1-1.2.3 Compatibility with Large Early Release Frequency Analyses

This Standard is not meant to be a replacement for the large early release frequency (LERF) portion of ASME/ANS RA-S-1.1-2024 [1-1] and associated upgrades of that Standard. Rather, this Standard extends beyond the LERF requirements of ASME/ANS RA-S-1.1-2024 [1-1] to include a more refined, comprehensive and realistic analysis of the full spectrum of possible radionuclide releases resulting from postulated severe accidents. Use of the Level 2 requirements provides a means of distributing the core damage frequencies (CDFs) determined via Level 1 PRA analyses into a set of RCs spanning the entire range of fission product release characteristics.

A subset of RCs represent large early releases, which have the potential for significant off-site early health effects. ASME/ANS RA-S-1.1-2024 [1-1] includes requirements for estimating the frequency of large early releases as a surrogate release metric for many PRA applications. Performing a full Level 2 analysis in accordance with this Standard provides an opportunity for a refined determination of LERF as a result of the greater

degree of modeling detail compared to that of the LERF evaluation prescribed in ASME/ANS RA-S-1.1-2024 [1-1].

The LERF technical element of ASME/ANS RA-S-1.1-2024 [1-1] remains an appropriate reference for LERF estimates. However, the completion of a Level 2 PRA according to this Standard allows the analyst to generate a refined estimate of LERF that is expected to remain consistent with the Capability Categories outlined in ASME/ANS RA-S-1.1-2024 [1-1] as applicable.

1-1.3 STRUCTURE FOR PRA REQUIREMENTS

1-1.3.1 PRA Technical Elements

The technical requirements for the PRA model are organized by their respective PRA technical elements. The PRA technical elements define the scope of the analysis of this Standard. This Standard specifies technical requirements for the PRA technical elements listed in Table 1-1.3-1.

1-1.3.2 High Level Requirements

A set of objectives and High Level Requirements (HLRs) is provided for each PRA technical element in the Technical Requirements section of Part 2 of this Standard. The HLRs set forth the minimum requirements for a technically acceptable baseline Level 2 PRA, independent of a PRA application. All HLRs are written by using “shall.” The HLRs are defined in general terms and present the overarching context for the derivation of more detailed SRs. The general terms used for HLRs represent not only the diversity of approaches that have been used to develop the existing Level 2 PRAs but also the need to accommodate future technological innovations.

1-1.3.3 Supporting Requirements

A set of SRs is stated for each HLR (that is included for each PRA technical element) in the Technical Requirements section in Part 2 of this Standard. All SRs are written by using “action verbs” rather than “shall.” The meaning of each action verb used in this Standard is stated in Nonmandatory Appendix (NMA) 1-A.

This Standard is intended for a wide range of PRA applications that require a corresponding range of PRA capabilities. PRA applications vary with respect to which risk metrics are employed, which decision criteria are used, the extent of reliance on the PRA results in supporting a decision, and the degree of resolution required for the factors that determine the significance of the subject of the decision. In developing the different portions of the PRA model, it is recognized that not every item will require the same level of detail, the same degree of plant specificity, or the same degree of realism.

Although the capabilities required for each portion of the PRA to support a PRA application fall on a continuum, two levels are defined and labeled Capability

Category I (CC-I) and Capability Category II (CC-II), so that requirements can be developed and presented in a manageable way. Table 1-1.3-2 describes, for three principal attributes of PRA, the bases for defining the Capability Category. This table was used to develop the SRs for each HLR.

The delineation of the Capability Categories within the SRs is generally aligned such that the degree of scope and level of detail, the degree of plant-specificity, and the degree of realism increase from CC-I to CC-II. The level of conservatism would generally tend to decrease as the Capability Category increases and more detail and more realism are introduced into the analysis. This is stipulated for many CC-I treatments; however, this is not true for all requirements and should not be assumed.

An example might be the treatment of hydrogen distribution and combustion within a large dry containment. One might propose that a “conservative” estimate of the load generated due to hydrogen combustion could be made by calculating the pressure generated from the complete combustion of a hydrogen mass representing oxidation of 100% of the Zircaloy cladding in the core. If this mass is assumed to be uniformly distributed within the containment free volume, the resulting flammable gas concentration might be at or below the lower flammability limit, and the resulting pressure increment might be very small. However, if a more refined spatial treatment of hydrogen transport and mixing within the containment is considered, very high concentrations might be estimated in small local regions of the containment that, if ignited, could threaten containment integrity. This example illustrates how a conservative simplifying assumption (100% Zircaloy oxidation) may

not be sufficient to produce a bounding result without more detailed and realistic analysis.

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. While all portions of the PRA need not have the same capability, the PRA model should be coherent. The SRs have been written so that within a Capability Category, the interfaces between portions of the PRA are coherent (e.g., the requirements for CETs are consistent with the definition of plant damage states).

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

For each SR, the minimum requirements necessary to meet CC-I and CC-II are defined. Some SRs apply to only 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 Categories is made in other associated SRs.

The Technical Requirements section of Part 2 of this Standard also specifies the required documentation to ensure traceability of the analysis.

Table 1-1.3-1 PRA Technical Elements Addressed by This Standard

Hazard Type/Group	Plant Operating States	PRA Technical Elements
All hazard types and groups	All plant operating states (not including Spent Fuel Pool)	Level 1/Level 2 PRA Interface (L1) Containment Performance Analysis (CP) Severe Accident Progression Analysis (SA) Probabilistic Treatment of Event Progression and Source Terms (PT) Source Term Analysis (ST) Evaluation and Presentation of Results (ER)

Table 1-1.3-2 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, and maintenance are modeled	Resolution and specificity are sufficient to identify the operating modes, initiating events, unmitigated system failures, system operating characteristics, mechanisms of containment failure, and severe accident progression phenomena that contribute to the significant accident progression sequences [Notes (1) and (2)].	Resolution and specificity are sufficient to identify the operating modes, initiating events, significant system failures, system operating characteristics, mechanisms of containment failure, and severe accident progression phenomena that contribute to the significant accident progression sequences [Note (2)].
2. Plant specificity: The degree to which plant-specific information is incorporated in modeling the as-built, as-operated plant	Use of generic data/models is acceptable except for the need to account for unique design and operational features of the plant that have bearing on the assessment of containment failure and release classes.	Plant-specific data/models are used for the significant contributors to the extent feasible.
3. Realism: The degree to which realism is incorporated in modeling the expected response of the plant	Departures from realism may have a moderate impact on the conclusions and risk insights as supported by state of the practice [Note (3)].	Departures from realism will have a small impact on the conclusions and risk insights as supported by state of the practice [Note (3)].

NOTES:

- (1) In this context “unmitigated systems failures” refers to failures of active or passive structures, systems, or components (SSCs) (including building structures) that are not restored or mitigated after the onset of core damage.
- (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 rather that they include 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 between moderate and 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 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 and/or tools were contemplated during the development of these requirements (for example, the use of tools such as MELCOR [1-4] or Modular Accident Analysis Program (MAAP) [1-5], were contemplated during the development of some requirements). Alternative methods and approaches or newly developed methods (NDMs) for meeting the requirements of this Standard may be used if they meet the HLRs and SRs presented in this Standard. Requirements for NDMs are provided in [Section 1-7](#). The requirements for the documentation of any particular method used are established in the documentation HLRs for each technical element, and requirements for peer review are described in [Section 1-6](#). All notes and commentaries, which follow many SRs, are nonmandatory. In addition, any example in the SR body or any NMA or note is not to be considered the only way to address an SR.

1-1.4 APPLICABILITY OF PRA TECHNICAL ELEMENTS

The use of a PRA and the Capability Categories that are required to be met for each of the PRA technical elements will differ among PRA applications. [Section 1-3](#) describes the activities to determine whether a PRA has the capability to support a specific PRA application of risk-informed decision-making (RIDM). Two different PRA Capability Categories are described in [Section 1-1.3](#). PRA capabilities are evaluated for each associated SR, rather than by specifying a Capability Category for specific parts or the whole PRA.

1-1.5 PRA CONFIGURATION CONTROL

[Section 1-5](#) states requirements for configuration control of a PRA (i.e., maintaining and upgrading a plant-specific PRA) such that the PRA represents the as-built, as-operated facility to a degree sufficient to support the PRA application for which it is used.

1-1.6 PEER REVIEW REQUIREMENTS

[Section 1-6](#) states the general requirements for a peer review to determine if the methods and its implementation in the PRA meet the requirements of the Technical Requirements section of each respective Part of this Standard.

1-1.7 ADDRESSING MULTIPLE HAZARD GROUPS

The approaches to modeling the plant damage resulting from different hazard groups vary in terms of the degree of realism and the level of detail achievable. For example, there are uncertainties that are unique to the

modeling of the different hazards and their effects 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.

For many PRA applications, it is necessary to include the combined impact on risk from those hazard groups for which it cannot be demonstrated that the impact on the decision being made is not risk insignificant. This can be done by using a single model that combines the PRA models for the different hazard groups or by combining the results from separate models. In either case, when combining the results from the different 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.

In some cases, the requirements for developing a Level 2 PRA model in [Part 2](#) refer to the requirements of ASME/ANS RA-S-1.1-2024 [1-1]. The requirements of ASME/ANS RA-S-1.1-2024 [1-1] should be applied to the extent needed, given the context of the modeling of each hazard group. In [Part 2](#), many of the requirements that differentiate between Capability Categories, either directly or by incorporating the requirements of ASME/ANS RA-S-1.1-2024 [1-1], do so on the basis of the analysis of significant contributors and significant accident progression sequences and/or cutsets for the hazard group being addressed. Because, as discussed above, there are differences in the way the PRA models for each specific hazard groups are developed; the requirements are best analyzed separately in a self-contained manner for each hazard.

Additionally, from a practical standpoint, PRA models are generally developed on a hazard group basis (e.g., a fire PRA, a seismic PRA, a high wind PRA). While they may be integrated into a single model with multiple hazards, the development is done on a hazard group basis. In CC-II, this Standard strives to ensure that the more significant contributors to each hazard group are understood and analyzed with an equivalent level of resolution across applicable SRs, plant specificity, and realism to not 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).

1-1.8 SCREENING CRITERIA

Note that this Standard does not include specific screening criteria that are not already described in the text of an individual SR.

1-1.9 UNDERSTANDING SIGNIFICANCE

One of the main outcomes of a state-of-practice PRA is the possibility to identify significant contributors based on quantitative criteria (i.e., an item under consideration that contributes above a certain percentage to an overall risk metric). Depending on the intended application of the Level 2 PRA, the identification of significant elements (or sequences) of the Level 2 PRA can be used to inform containment design and improve severe accident management strategies. Level 2 items identified as significant are also prime target for analysis refinements, aimed at enhancing the realism of the associated insights. The requirements provided in this Standard ensure that the analysis maintains an appropriate level of completeness and that even at CC-I it is possible to identify significant contributors.

ASME/ANS RA-S-1.1-2024 [1-1] uses the concept “risk-significant” to define those components and actions where realistic modeling is necessary to meet CC-II, where risk-significance is defined with respect to the Level 1 risk metrics of CDF and LERF. In a Level 1 PRA these surrogate metrics are defined in regulation, where CDF is generally regarded as a surrogate for the individual latent cancer fatality risk quantitative health objective (QHO) and LERF has been shown to be an adequate surrogate for the individual early fatality risk QHO. This Standard allows use of multiple surrogate release metrics to characterize the Level 2 end states. The radiological releases estimated in the Level 2 PRA may be propagated into a Level 3 PRA, which quantifies the consequences and impacts of the releases on the environment and public. As the Level 2 metrics were not necessarily considered stand-alone measures of the risk (i.e., frequency and consequence) of containment releases, this Standard defines specific “significant” metrics as targets for refinement and realism within individual SRs. Therefore, while ASME/ANS RA-S-1.1-2024 [1-1] may define “risk-significant” with respect to the LERF surrogate release risk metric, this Standard uses the more general term “significant” when identifying contributors to all release classes and containment states. Use of term “risk-significant” may be appropriate in Level 2 for the respective surrogate release metrics where there are consistent assumed consequences [e.g., LERF or large release frequency (LRF)] depending on the needs of an application.

Within the Level 2 PRA, the term “significant” defines how much realism is necessary to meet CC-II of some SRs and the importance of those contributors to the Level 2. They are NOT intended to be definitions of what is significant in a particular application nor to provide a direct metric on radiological consequences. Indeed, in the context of a specific application, they may be either too loose or too restrictive, depending on what is being evaluated. In the context of this Standard, the decisions on applying these definitions and/or defining

what is significant to a decision would be addressed in [Section 1-3](#).

While the Level 2 Standard has chosen to not use the term “risk-significant,” it does retain the formalism in identifying and establishing the concept of significance. Therefore, significant contributors in this Standard are identified in a parallel fashion to the risk-significant contributors in ASME/ANS RA-S-1.1-2024 [1-1] based on quantitative criteria (i.e., an item under consideration that contributes above a certain percentage to the overall risk as measured by the metrics under consideration). In that context, the Level 2 concept of significance is established with respect to the percent contribution to a given metric. It is recognized that “risk” is a product of frequency and consequence, and this concept could be applied to any metric. However, since not all Level 2 release measures can be assumed to have the same consequences, it was determined that significance should be determined with respect to each metric defined; thus, the term “significance” is used in place of “risk-significance” in this Standard.

In line with the above, [Table 1-1.9-1](#) specifies the quantitative criteria generally to be used in determining significance for the various modeling items (contributors). If these quantitative criteria are not used, justification for any alternative quantitative criteria shall be documented. The documentation shall describe how the alternative quantitative criteria meet the intent of the criteria in [Table 1-1.9-1](#). These alternative quantitative criteria shall be peer reviewed for their appropriateness and ability to adequately determine significance such that the integrity of the PRA model is maintained. Once the potential significant contributors are identified along with the specific Technical Requirements, the next major step is to apply the needed refinements into the modeling inputs. The Level 2 PRA applies significance ranking to the following:

- (a) significant accident progression sequences
- (b) significant basic events
- (c) significant containment challenges
- (d) significant contributors
- (e) significant release categories

When Level 2 PRA is not used as direct input to a Level 3, Level 2 metrics typically reflect intermediate measures that are intended to characterize the Level 2 model contributors and are not a direct measure of risk. In these circumstances, the metrics that are to be used for assessment of significance shall be selected; however, the various interim metrics are not necessarily equally significant to public risk. These interim states can be important in establishing the adequacy of the Level 2 PRA. Specifically, significance measures can be used in the iteration process to determine the level of detail necessary for a Level 2 PRA model. Hence, initial simplifying assumptions that may impact multiple portions of the PRA model may need to be reviewed and modified to

increase the PRA model realism. 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. When a radiological consequence is used as a specific metric, dose and public impact assessments need to be considered. These features are not specifically considered in the Level 2 PRA. Therefore, actual

risk significance of Level 2 PRA accident progression sequences, basic events, and other relevant items, may not be able to be established solely within the Level 2 PRA and could be defined iteratively using the calculated consequences derived as part of the Level 3 PRA.

Table 1-1.9-1 describes how significance is determined for the different types of modeling items (contributors).

Table 1-1.9-1 Significance Determination

Item	Criteria for Significance Determination [Note 1]
Significant accident progression sequence	One of the set of accident sequences contributing to the governing surrogate release metric (such as LERF, LRF, or an alternate release metric for a specific application) resulting from the analysis of a specific hazard group that, when rank-ordered by decreasing frequency, sum to a specified percentage of the surrogate release metric or that individually contribute more than a specified percentage of the surrogate release metric for that hazard group. The summed percentage of 95% and the individual percentage of 1% of the surrogate release metric are generally used.
Significant basic event	This basic event contributes significantly to the computed risks for a specific hazard group. This contribution generally includes any basic event that has a Fussell-Vesely (FV) importance greater than 0.005 or a risk achievement worth (RAW) importance greater than 2 relative to the governing surrogate release metric (such as LERF, LRF, or an alternate release metric for a specific application).
Significant containment challenge	A containment challenge that results in a containment failure mode that is represented in a significant accident progression sequence or is represented by a significant basic event.
Significant contributor	A contributor that is an essential characteristic (e.g., containment failure mode, physical phenomena, or basic event) of a significant accident progression sequence, release category, or governing surrogate release metric (such as LERF, LRF, or an alternate release metric for a specific application).
Significant release category	One of the set of radionuclide release categories contributing to the governing surrogate release metric (such as LERF, LRF, or an alternate release metric for a specific application) that, when rank-ordered by decreasing frequency, sums to 95% of the surrogate release metric or overall release frequency (excluding design basis leakage RCs) or individually contributes more than 1% of the surrogate release metric or 5% of the overall release frequency.

NOTE:

- (1) If these criteria are not used, justification for any alternative criteria shall be documented. The documentation shall describe how the alternative criteria meet the intent of the stated criteria in this table. These alternative criteria shall be peer reviewed for their appropriateness and ability to adequately determine significance such that the integrity of the PRA model is maintained.

Section 1-2

Acronyms and Definitions

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

1-2.1 ACRONYMS

<i>ABWR</i> : advanced boiling water reactor	<i>LERF</i> : large early release frequency
<i>ANS</i> : American Nuclear Society	<i>LOCA</i> : loss of coolant accident
<i>AOP</i> : abnormal operating procedure	<i>LOOP</i> : loss of off-site power
<i>APWR</i> : advanced pressurized water reactor	<i>LPI</i> : low pressure injection
<i>ASME</i> : American Society of Mechanical Engineers	<i>LPSD</i> : low power and shutdown
<i>BMMT</i> : basemat melt-through	<i>LR</i> : large release
<i>BWR</i> : boiling water reactor	<i>LRF</i> : large release frequency
<i>BWROG</i> : Boiling Water Reactor Owners Group	<i>LWR</i> : light water reactor
<i>BWST</i> : borated water storage tank	<i>MAAP</i> : Modular Accident Analysis Program
<i>CC-I and CC-II</i> : Capability Categories I and II	<i>MOV</i> : motor-operated valve
<i>CCF</i> : common cause failure	<i>NDM</i> : newly developed method
<i>CDF</i> : core damage frequency	<i>NMA</i> : Nonmandatory Appendix
<i>CET</i> : containment event tree	<i>NRC</i> : Nuclear Regulatory Commission
<i>CST</i> : condensate storage tank	<i>PDS</i> : plant damage state
<i>DDT</i> : deflagration to detonation transition	<i>POS</i> : plant operating state
<i>DW</i> : drywell	<i>PRA</i> : probabilistic risk assessment
<i>EOP</i> : emergency operating procedure	<i>PSA</i> : probabilistic safety assessment
<i>EP</i> : emergency planning	<i>PWR</i> : pressurized water reactor
<i>EPRI</i> : Electric Power Research Institute	<i>QHO</i> : quantitative health objective
<i>ESBWR</i> : economic simplified boiling water reactor	<i>RAW</i> : risk achievement worth
<i>FSG</i> : FLEX support guideline	<i>RC</i> : release category
<i>FV</i> : Fussell-Vesely importance measure	<i>RCS</i> : reactor coolant system
<i>HEP</i> : human error probability	<i>RIDM</i> : risk-informed decision-making
<i>HFE</i> : human failure event	<i>RPV</i> : reactor pressure vessel
<i>HLR</i> : High Level Requirement	<i>RWST</i> : refueling water storage tank
<i>HPME</i> : high pressure melt ejection	<i>SAG</i> : severe accident guideline
<i>HRA</i> : human reliability analysis	<i>SAMG</i> : severe accident management guideline
<i>HVAC</i> : heating, ventilation, and air conditioning	<i>SG</i> : steam generator
<i>HW</i> : high wind	<i>SGTR</i> : steam generator tube rupture
<i>IAEA</i> : International Atomic Energy Agency	<i>SR</i> : Supporting Requirement
<i>ISLOCA</i> : interfacing systems loss of coolant accident	<i>SRV</i> : safety relief valve
	<i>SSC(s)</i> : structure(s), system(s), or component(s)
	<i>THERP</i> : Technique for Human Error Rate Prediction (see NUREG/CR-1278 [1-6])
	<i>TSC</i> : technical support center

1-2.2 DEFINITIONS

accepted method: a method that the regulatory body has used or accepted for the specific risk-informed application for which it is proposed.

accident progression framework: a logic model that accounts for the possible pathways and outcomes with respect to the physical progression of a core damage accident and the containment response. The accident progression framework may be constructed in different ways.

accident progression sequence: a unique combination of events that clearly delineate the chronological and physical progression of core damage, containment response, and fission product release to the environment.

accident sequence: a representation in terms of an initiating event followed by a sequence of failures or successes of events (e.g., system, function, or operator performance) that can lead to undesired consequences, with a specified end state (e.g., core damage or large early release).

aleatory uncertainty: the uncertainty inherent in a non-deterministic (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”).

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-to-be-built, and as-to-be-operated” plant.)

assumption: a judgment that is made in the development of the PRA model either for modeling convenience or because of 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 individual modeled components under the assumption that the consequence of any individual combination of components is the same.

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

at-power: 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.

beneficial failure: a failure of an active or passive structure, system, or component (SSC) that alters accident progression in a manner that reduces the severity of the reactor or containment status or mitigates the consequences of subsequent events. An example of a beneficial failure would be creep rupture of the reactor coolant system (RCS) hot leg prior to failure of other RCS components that would have a more severe consequence [i.e., consequential steam generator tube rupture (SGTR) or high-pressure failure of the reactor pressure vessel (RPV) lower head].

bridge tree: an event tree (or equivalent logic structure) that extends the sequences delineated in the Level 1 PRA to account for the status of containment systems. A bridge tree is sometimes used to link (or provide a “bridge” between) the Level 1 event trees (or equivalent) for core-damage sequences and the Level 2 containment event tree, especially when the latter is constructed solely to reflect the potential severe accident phenomena.

Capability Category: see [Table 1-1.3-2](#)

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 deflagration to detonation transition point).

common cause failure: 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.

component: an item in a nuclear power plant, such as a vessel, pump, valve, or circuit breaker.

consensus method/model: a method or model that the regulatory body has used or accepted for the specific risk-informed application for which it is proposed.

conservative: use of information (e.g., assumptions) such that the assessed outcome is meant to be less favorable than the expected outcome. With respect to Level 2 PRA, this involves frequency of release categories as well as the timing and magnitude of the radiological release (i.e., source term). Sensitivity studies may be used to confirm the assumed bias has produced the intended outcome.

containment bypass: a direct or indirect flow path that may allow the release of radioactive material directly to the environment, bypassing the containment.

containment challenge: severe accident conditions (e.g., plant thermal hydraulic conditions or phenomena) that may result in compromising containment integrity. These conditions or phenomena can be compared with containment capability to determine whether a containment failure mode results.

containment event tree: a logic diagram that graphically represents the status of the containment and containment equipment when subjected to severe accident loads. In a PRA, a CET begins with the onset of core damage and progresses through a limited number of branches that depict the various scenarios of the containment and containment equipment performance when subjected to severe accident loads (e.g., high temperatures, pressures).

containment failure: loss of integrity of the containment pressure boundary from a core damage accident that results in unacceptable leakage of radionuclides to the environment.

containment failure mode: the manner in which a containment radionuclide release pathway is created. It encompasses both the structural failures of containment induced by containment challenges when they exceed containment capability and the failure modes of containment induced by human failure events (HFEs), isolation failures, or bypass events such as interfacing systems loss of coolant accident (ISLOCA).

containment performance: a measure of the response of a nuclear plant containment to severe accident conditions.

core damage: uncovering and heat-up of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in off-site public health effects.

core damage frequency: expected number of core damage events per unit of time.

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

end state: the set of conditions at the end of an accident 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), plant damage states for Level 1 PRA sequences, and release categories for Level 2 PRA sequences (including those contributing to LERF).

epistemic uncertainty: the uncertainty attributable to incomplete knowledge about a phenomenon that affects our ability to model it. Epistemic uncertainty is reflected in 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.

equipment: a term used to broadly cover the various components in a nuclear power plant. 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.

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 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.

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.

external hazard: a hazard originating outside a nuclear power plant that directly or indirectly causes an initiating event and may cause safety system failures or operator errors that may lead to core damage or large early release. Hazards such as earthquakes, tornadoes, and floods from sources outside the plant and fires from sources outside the plant are considered external hazards. (See also *internal event*.) By historical convention, a loss of off-site power (LOOP) not caused by another external hazard is considered an internal event.

facilitator/integrator: a single entity (individual, team, company, etc.) that 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).

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

failure rate: 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.

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, flood level, or high pressure or temperature loads on containment. 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 modeling and data uncertainty in the median capacity.

Fussell-Vesely: for a specified basic event, Fussell-Vesely (FV) importance is the fractional contribution to the total of a selected figure of merit for all accident sequences containing that basic event. For PRA quantification methods that include nonminimal cutsets and success probabilities, the FV 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.

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 loss of coolant accidents (LOCAs) and LOOPs. In some cases, a hazard group may consist of only one hazard (e.g., the seismic hazard), in which case the hazard and the hazard group are considered to be synonymous.

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 flood, seismic, and high winds (HW). In some cases, a hazard group may only consist of one hazard (e.g., the seismic hazard), in which case the hazard group and the hazard are considered to be synonymous.

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.

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

human error probability: 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 human error probability (HEP) is the probability of the human failure event (HFE).

human failure event: 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.

initiating event: a perturbation of the steady-state operation of the plant that challenges plant control and safety systems whose failure could potentially lead to core damage. An initiating event is defined in terms of the change in plant status that results in a condition requiring a reactor trip (e.g., loss of main feedwater system, small LOCA) or a manual trip prompted by conditions other than those in the normal shutdown procedure when the plant is at-power. An initiating event may result from human causes, equipment failure from causes internal to the plant (e.g., hardware faults, floods, or fires) or external to the plant (e.g., earthquakes or HWs), 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.) that 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.

interfacing systems LOCA: a LOCA when a breach occurs in a system that interfaces with the RCS, where isolation between the breached system and the RCS fails. An ISLOCA is usually characterized by the overpressurization of a low-pressure system when subjected to RCS pressure and can result in containment bypass.

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 core damage.

large early release: a large release (LR) occurring before the effective implementation of off-site emergency response and protective actions such that there is a potential for early health effects.

large early release frequency: expected number of large early releases per unit of time.

large release: the release of airborne fission products to the environment such that there are significant off-site impacts. LR and significant off-site impacts may be defined in terms of quantities of fission products

released to the environment, status of fission product barriers and scrubbing, or dose levels at specific distances from the release, depending on the specific analysis objectives and regulatory requirements.

large release frequency: expected number of LRs per unit of time.

Level 1 analysis: identification and quantification of the sequences of events leading to the onset of core damage.

Level 2 analysis: identification and quantification of the sequences of events that impact the reactor and containment response, starting with the onset of core damage and leading to the point of radioactive release to environment or the determination that a release will not occur.

Level 2 PRA: A PRA that encompasses the Level 1 and Level 2 analyses. See also *Level 2 analysis*.

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

low power: a plant operating state (POS) (or set of POSs) during which the reactor is at reduced power below nominal full-power conditions. In these POSs, the power level may be changing as the reactor is shutting down or starting up, 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 (e.g., taking manual control of feedwater level).

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

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 for which a system or component is required to operate in order to successfully perform its function.

model: a qualitative and/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.

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.

parameter uncertainty: the uncertainty in the value of an input parameter that represents the degree of belief in the range of values the input parameter may assume. Examples of parameter uncertainty include, but are not limited to, probability distributions or confidence intervals (i.e., a range of probability values within which the actual value of the input parameter is expected to reside) for an input parameter such as an initiating event frequency or a component failure probability.

passive SSC: an SSC that performs one or more safety functions either fully or partially via passive means (i.e., relying on natural physical processes such as natural convection, thermal conduction, radiation, gravity, or pressure differentials, or depending on the integrity of a pressure boundary or structural component). Examples include piping systems that are used to maintain an inventory of fluid and deliver flow along a fluid path, and structural supports for SSCs.

phenomenological event: an observable event that occurs if the governing physical and chemical phenomena proceed in a particular but possibly uncertain way. Such events are typically defined within the context of known (or assumed) initial and boundary conditions concerning the status of SSCs and the actions of the operating crew. Uncertainties in such processes or events are typically governed by epistemic uncertainty in governing processes or in the fidelity of analytical models to accurately calculate the behavior of known physical/chemical processes.

plant: a general term used to refer to a nuclear power facility (e.g., "plant" could be used to refer to a single unit or multi-unit site).

plant damage state: group of accident sequence end states that have similar characteristics with respect to accident progression and containment or engineered safety feature operability.

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. POS is a basic modeling device used for a phased-mission risk assessment that discretizes the plant conditions for specific phases of a low power and shutdown (LPSD) evolution. Examples of such plant conditions include core decay heat level, primary water level, primary temperature, primary vent status, containment status, and decay heat removal mechanisms. Examples of risk impacts that are dependent on POS definition include the selection of initiating events, initiating event frequencies, definition of accident sequences, success criteria, and accident sequence quantification.

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

point estimate: estimate of a parameter in the form of a single number.

PRA application: a documented analysis based in part or whole on a plant-specific PRA that is used to assist in decision-making with regard to the design, licensing,

procurement, construction, operation, or maintenance of a nuclear power plant.

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 that were not previously included within the PRA, an implementation of a PRA method in a different context, or the incorporation of a method not previously used.

probabilistic risk assessment: a quantitative assessment of the risk including all technical elements for modeled hazards associated with plant operation and maintenance that is measured in terms of frequency of occurrence of risk metrics, such as core damage or a radioactive material release and its effects on the health of the public (also referred to as a “probabilistic safety analysis”).

radionuclide group: a set of radionuclides that are treated as a single representative species for the purpose 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. A delineation of radionuclide groups used in many severe accident computational models can be found in NUREG-1465 [1-7].

radionuclide release category: see *release category*.

realism: an accurate representation (to the extent practical) of the expected response of the as-built, as-operated 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 (HRA) techniques.

release category: a group of accident progression sequences that would generate a similar source term to the environment. Similarity in this context depends on the level of fidelity of the analysis and the number of release categories (RCs) used to span the entire spectrum of possibilities. Similarity is generally measured in terms of the overall (cumulative) release of activity to the environment, the timing of the release and (in certain applications) other physical characteristics of the source term.

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.

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.

risk: probability 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?
- (c) What are the consequences if it occurs?

risk achievement worth 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.

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

safety function: function that must be performed to control the sources of energy in the plant and radiation hazards.

safety systems: those systems that are designed to prevent or mitigate a design-basis accident.

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 accident sequence or its consequences.

severe accident: an accident that involves extensive core damage and fission product release into the reactor vessel and containment, with potential release to the environment.

severe accident management guidelines: guidelines developed to provide steps that can be taken to mitigate accident progression after transition from the emergency operating procedures because of more severe conditions (e.g., core damage).

shall: used to state a mandatory requirement.

should: used to state a recommendation.

shutdown: the collection of POSs during which the reactor is subcritical. This term is interchangeable with the term “outage.”

significant accident progression sequence: see definitions in [Table 1-1.9-1](#).

significant basic event: see definitions in [Table 1-1.9-1](#).

significant containment challenge: see definitions in [Table 1-1.9-1](#).

significant contributor: see definitions in [Table 1-1.9-1](#).

significant release category: see definitions in [Table 1-1.9-1](#).

source of model uncertainty: the uncertainty associated with the variability of an input of interest where the input of interest can be derived or calculated via different modeling approaches, where the selected approach is not clearly more correct or does not represent a consensus of

the technical community, and where the choice of modeling approach is known to have an impact 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).

source term: the characteristics of a radionuclide release at a particular location including the physical and chemical properties of released material, release magnitude, heat content (or energy) of the carrier fluid, location relative to local obstacles that would affect transport away from the release point, and the temporal variations in these parameters (e.g., time of release, duration, etc.).

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.

state-of-knowledge correlation: the correlation that arises between sample values when performing uncertainty analysis for cutsets consisting of basic events by using a sampling approach (e.g., the Monte Carlo method). When the state of knowledge correlation is included, it results, for each sample, in the same value being used for all basic event probabilities to which the same data apply.

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

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.

surrogate release metric: a grouping of RCs with defined attributes that are assumed to have similar or bounding

consequences [e.g., those used to support the Nuclear Regulatory Commission's (NRC's) quantitative health objective (QHOs)] and may be used for specific applications where a full Level 3 PRA is not utilized. Examples of surrogate release metrics include LERF, LRF, or an alternate release metric for a specific application.

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

termination time: time following a severe accident at which the radionuclide release calculation is terminated (truncated) for Level 2 analysis purposes.

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.

top event: 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.

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 accident sequences/cutsets, system-level cutsets, and sequence/cutset database retention).

unavailability: the probability that a system or component is not capable of supporting its function including, but not limited to, the time it is disabled for test or maintenance. 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.

unreliability: the probability that a system or component will not perform its specified function under given conditions on demand or for a prescribed time.

Section 1-3

PRA Scope and Capabilities in Support of Risk-Informed Applications

1-3.1 PURPOSE

This Section describes needed activities to establish the capability of a PRA to support a particular risk-informed application. For this Section, the term “PRA” (or “PRA model”) can refer to either an integrated model that includes all relevant hazard groups or multiple PRA models that address one or more hazard groups. For a specific application, PRA capabilities are evaluated in terms of Capability Categories for individual 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 [Part 2](#) of this Standard. The process is intended to be used with PRAs that have had a peer review that meets the requirements of the Peer Review Section of each respective Part of this Standard.

While this Standard may be used to establish more realistic estimates of LERFs, meeting this Standard does not necessarily imply regulatory acceptance of the LERF estimates generated from this Standard for specific applications that use ASME/ANS RA-S-1.1-2024 [1-1]. Risk-informed applications using this Standard are expected to focus on applications that (1) require environmental fission product release estimates as input to a Level 3 assessment or (2) where a conditional containment failure metric and/or release class frequency specific target(s) are used to establish the effectiveness of release mitigation strategies.

1-3.2 IDENTIFICATION OF APPLICATION AND DETERMINATION OF CAPABILITY CATEGORIES

1-3.2.1 Identification of Application

Define the application by

- (a) evaluating the plant design or operational change being assessed
- (b) identifying the SSCs and plant activities affected by the change including the cause-effect relationship between the plant design or operational change and the PRA model
- (c) identifying the hazard groups, PRA model scope, and PRA risk metrics that are needed to assess the change

EPRI TR-105396 [1-8] and RG 1.174 [1-9] provide guidance for the above activities.

1-3.2.2 Determination of Capability Categories

Other Parts of this Standard state SRs for the PRA Capability Categories whose attributes are described in [Section 1-1.3](#).

For many of the SRs, the distinction between Capability Categories is based on the treatment of significant contributors. Definitions in this Standard containing the word “significance” or “significant” are generally written from the perspective of a specific hazard group. It is important to recognize that, for applications whose risk stems from more than one hazard group, these definitions should be generalized to apply to the sum of risks from all contributing hazard groups. “Significance” should also be treated differently for those SRs that refer to SRs in other hazard groups.

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

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

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

Section 1-4

Requirements for Use of Expert Judgment

1-4.1 PURPOSE

This Section states general requirements for use of expert judgment.

1-4.2 USE OF EXPERT JUDGMENT

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

Guidance from NUREG/CR-6372 [1-10] and NUREG 1563 [1-11] may be used to meet the requirements in this paragraph. Other approaches, or a mix of these, may also be used.

1-4.2.1 Objective of Using Expert Judgment

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

1-4.2.2 Identification of the Technical Issue

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

1-4.2.3 Determination of the Need for Outside Expert Judgment

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

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

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

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

(c) Judgments are required to assess whether bounding assumptions or calculations are appropriately conservative.

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

1-4.2.4 Identification of Expert Judgment Process

The PRA analysis team shall determine

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

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

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

1-4.2.5 Identification and Selection of Evaluator Experts

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

(a) their own input

(b) their representation of the community distribution

1-4.2.6 Identification and Selection of Technical Issue Experts

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

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

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

1-4.2.7 Responsibility for the Expert Judgment

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

Section 1-5

PRA Configuration Control Program

1-5.1 INTRODUCTION

This Section states requirements for a configuration control program to support the use of a PRA in risk-informed decisions for nuclear power plants. The HLRs and SRs for this PRA Configuration Control (CC) Program are contained in [Table 1.5.3-1](#), [Table 1.5.3-2](#), [Table 1.5.3-3](#), [Table 1.5.3-4](#), [Table 1.5.3-5](#), and [Table 1.5.3-6](#). As these are administrative requirements, there is no gradation across Capability Categories. A discussion of the requirements is presented below.

1-5.2 OBJECTIVE

The objective of the configuration control program is to ensure that when a PRA is to be used in risk-informed decisions, it represents the as-built, as-operated plant at the time of the decision. Furthermore, it ensures that any updates of the PRA are consistent with the Technical Requirements of this Standard.

Table 1-5.3-1 High Level Requirements for PRA Configuration Control (CC) Program

Designator	Requirement
HLR-CC-A	The PRA Configuration Control Program shall include a process for monitoring changes to the plant design, operation, PRA technology, and industry experience and for collecting updated performance information that could result in changes to PRA inputs.
HLR-CC-B	The PRA Configuration Control Program shall include a process that maintains and upgrades the PRA to be consistent with the as-built, as-operated plant.
HLR-CC-C	The PRA Configuration Control Program shall consider the cumulative impact of pending changes in the performance of risk applications.
HLR-CC-D	The PRA Configuration Control Program shall include a process that maintains configuration control of computer codes and associated files used to support PRA.
HLR-CC-E	The PRA Configuration Control Program and its implementation shall be documented.

Table 1-5.3-2 Supporting Requirements for HLR-CC-A

The PRA Configuration Control Program shall include a process for monitoring changes to the plant design, operation, PRA technology, and industry experience and for collecting updated performance information that could result in changes to PRA inputs ([HLR-CC-A](#)).

Index No. CC-A	Requirements
CC-A1	IMPLEMENT a process to track plant changes, PRA technology, and related industry equipment performance/operational experience focused on collecting the necessary information to update PRA inputs.
CC-A2	In the information collected, INCLUDE the plant-specific changes in design, operation, and maintenance of the plant that impact, for example, the following: (a) operating procedures and practices (e.g., operations orders) (b) emergency and abnormal operating procedures (c) design configuration (d) initiating event frequencies (e) system or subsystem unavailabilities (f) component failure rates (g) maintenance policies (h) operator training (i) technical specifications (j) engineering calculations (k) emergency plan (l) accident management programs
CC-A3	In the information collected, INCLUDE changes to external facilities, sources of external hazards, or internal or external features that impact how external hazards may affect the plant. Such information may include, but is not limited to the following: (a) changes in dam operating procedures that impact water release strategies (b) regional changes that impact riverine flooding hazard analysis (c) capabilities of external response centers if such centers are credited in the PRA
CC-A4	In the information collected, INCLUDE changes in industry experience that could impact the following: (a) estimation of initiating event frequencies (b) generic system or subsystem unavailabilities (c) generic component failure rates (d) initiating events
CC-A5	In the information collected, INCLUDE changes to the PRA technology that could change the results of the PRA model.

Table 1-5.3-3 Supporting Requirements for HLR-CC-B

The PRA Configuration Control Program shall include a process that maintains and upgrades the PRA to be consistent with the as-built, as-operated plant ([HLR-CC-B](#)).

Index No. CC-B	Requirements
CC-B1	EVALUATE changes in PRA inputs or new information identified pursuant to HLR CC-A to determine whether such information warrants PRA maintenance or PRA upgrade. INCLUDE in the PRA changes identified per HLR-CC-A .
CC-B2	INCLUDE in the PRA those maintenance or upgrade changes implemented per HLR-CC-A that would impact risk-significant insights.
CC-B3	PERFORM a peer review of portions of the PRA that are affected by a PRA upgrade in accordance with the applicable requirements specified in the Peer Review Section of each respective Part of this Standard. The scope may be limited within a technical element to only the SRs that are germane to a specific PRA upgrade.
CC-B4	ENSURE that changes to the PRA due to PRA maintenance or upgrade meet the requirements of the Technical Requirements section of each respective Part of this Standard.
CC-B5	REVIEW maintenance or upgrade changes made to the PRA by using a utility-approved process.

Table 1-5.3-4 Supporting Requirements for HLR-CC-C

The PRA Configuration Control Program shall consider the cumulative impact of pending changes in the performance of risk applications ([HLR-CC-C](#)).

Index No. CC-C	Requirements
CC-C1	IDENTIFY plant changes that have been identified to have a potential impact on PRA.
CC-C2	IDENTIFY known industry issues or events and PRA technology changes that may have an impact on the PRA model.

Table 1-5.3-5 Supporting Requirements for HLR-CC-D

The PRA Configuration Control Program shall include a process that maintains configuration control of computer codes used to support and perform PRA analyses ([HLR-CC-D](#)).

Index No. CC-D	Requirements
CC-D1	ENSURE that the computer codes and associated files used to support and to quantify the PRA are controlled to ensure consistent, reproducible results.

Table 1-5.3-6 Supporting Requirements for HLR-CC-E

The PRA Configuration Control Program and its implementation shall be documented ([HLR-CC-E](#)).

Index No. CC-E	Requirements
CC-E1	DOCUMENT the Configuration Control Program and the performance of the above elements in a manner adequate to demonstrate that the PRA is being maintained consistently with the as-built, as-operated plant. The documentation typically includes (a) a description of the process used to monitor PRA inputs and collect new information (b) evidence that the aforementioned process is active (c) descriptions of proposed and implemented changes (d) a description of changes in a PRA due to each PRA upgrade or PRA maintenance (e) a record of the performance and results of the appropriate PRA reviews (consistent with the requirements of Section 1-6.6) (f) a record of the process and results used to address the cumulative impact of pending changes (g) a description of the process used to maintain software configuration control (h) a record of the process and results used to evaluate changes on previously implemented risk-informed decisions
CC-E2	DOCUMENT the bases for the changes made to the PRA model.