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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The American Society of Mechanical Engineers

ANS

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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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(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

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(3) the Standard and the paragraph, figure, or table number

(4) the editions of the Standard to which the proposed case applies

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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

ABWR: advanced boiling water reactor

ANS: American Nuclear Society

AOP: abnormal operating procedure

APWR: advanced pressurized water reactor

ASME: American Society of Mechanical Engineers

BMMT: basemat melt-through

BWR: boiling water reactor

BWROG: Boiling Water Reactor Owners Group

BWST: borated water storage tank

CC-I and CC-II: Capability Categories I and II

CCF: common cause failure

CDF: core damage frequency

CET: containment event tree

CST: condensate storage tank

DDT: deflagration to detonation transition

DW: drywell

EOP: emergency operating procedure

EP: emergency planning

EPRI: Electric Power Research Institute

ESBWR: economic simplified boiling water reactor

FSG: FLEX support guideline

FV: Fussell-Vesely importance measure

HEP: human error probability

HFE: human failure event

HLR: High Level Requirement

HPME: high pressure melt ejection

HRA: human reliability analysis

HVAC: heating, ventilation, and air conditioning

HW: high wind

IAEA: International Atomic Energy Agency

ISLOCA: interfacing systems loss of coolant accident

LERF: large early release frequency

LOCA: loss of coolant accident

LOOP: loss of off-site power

LPI: low pressure injection

LPSD: low power and shutdown

LR: large release

LRF: large release frequency

LWR: light water reactor

MAAP: Modular Accident Analysis Program

MOV: motor-operated valve

NDM: newly developed method

NMA: Nonmandatory Appendix

NRC: Nuclear Regulatory Commission

PDS: plant damage state

POS: plant operating state

PRA: probabilistic risk assessment

PSA: probabilistic safety assessment

PWR: pressurized water reactor

QHO: quantitative health objective

RAW: risk achievement worth

RC: release category

RCS: reactor coolant system

RIDM: risk-informed decision-making

RPV: reactor pressure vessel

RWST: refueling water storage tank

SAG: severe accident guideline

SAMG: severe accident management guideline

SG: steam generator

SGTR: steam generator tube rupture

SR: Supporting Requirement

SRV: safety relief valve

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

THERP: Technique for Human Error Rate Prediction (see NUREG/CR-1278 [1-6])

TSC: 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 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").

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: uncovery 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.

Section 1-6
Peer Review

1-6.1 PURPOSE

This Section states requirements for peer review of the PRA to be used in risk-informed decisions for commercial nuclear power plants. Those portions of PRAs used for PRA applications applying this Standard shall be peer reviewed. The peer review shall assess the PRA to the extent necessary to determine whether the method and its implementation meet the requirements of this Standard. Another purpose of the peer review is to determine the potential gaps in the PRA relative to this Standard's requirements. The peer review need not assess all aspects of the PRA against all requirements in the Technical Requirements section of Part 2 of this Standard but must address all SRs relevant to the scope of the peer review. However, enough aspects of the PRA shall be reviewed for the reviewers to achieve consensus on the adequacy of the assessment of each applicable SR as well as on the methods and their implementation for each PRA technical element.

1-6.1.1 Documentation and Self-Assessment

The following are prerequisites for performing the peer review:

(a) The Level 2 PRA is based on a documented and peer-reviewed Level 1 PRA with all significant deficiencies relevant to the Level 2 PRA resolved for all applicable hazards, OR with clear identification of these deficiencies and documentation of their potential impacts on the Level 2 PRA to be reviewed.

(b) The Level 2 PRA has documented the supporting analyses/calculations, including the independent reviews performed.

(c) A self-assessment of the Level 2 PRA has been conducted to establish the extent to which the PRA meets the requirements of this Standard. The results of the self-assessment process shall be documented.

1-6.1.2 Scope

Peer reviews shall be performed against the requirements in Part 2 of this Standard that are applicable to the Level 2 PRA that is being used to support risk-informed decisions.

The scope of the peer review may be a "focused-scope" peer review. A focused-scope peer review is a subset of a complete (full-scope) peer review and involves specified SRs. A focused-scope peer review may be requested

(a) to support a specific application that does not involve the complete Level 2 PRA model

(b) to address changes to the PRA model as a result of upgrades, or

(c) to close significant deficiencies from previous peer reviews

When included in the scope of a peer review, an NDM shall be reviewed following the dedicated requirements discussed in Section 1-7.

1-6.1.3 Peer Review Process

The review shall be performed using a written process that assesses the requirements of the Technical Requirements section of Part 2 of this Standard and addresses the requirements in this Section.

The peer-review process shall consist of the following elements:

(a) selection of the peer-review team

(b) training in the peer-review process

(c) an approach to be used by the peer-review team for assessing if the PRA meets the supporting requirements of the Technical Requirements section in Part 2 of this Standard

(d) management and resolution of potential differing professional opinions

(e) documentation of the results of the review

1-6.2 PEER REVIEW TEAM COMPOSITION AND PERSONNEL QUALIFICATIONS

1-6.2.1 Collective Team

The peer-review team shall consist of personnel whose collective qualifications include

(a) the ability to assess all the PRA technical elements of the Technical Requirements of this Standard, as applicable, and the interfaces between those elements

(b) the collective knowledge of the plant nuclear steam supply system design, containment design, and plant operation

(c) knowledge of severe accident phenomenology relevant to the plant design, containment failure modes associated with severe accident challenges and source term characterization

(d) knowledge of severe accident methods applied in the performance of the Level 2 PRA

1-6.2.2 Individual Team Members

The peer-review team members individually shall be

(a) knowledgeable of the requirements in this Standard for their area of review

(b) experienced in performing the activities related to the PRA technical elements for which the reviewer is assigned

(c) independent from the team that developed the PRA model or the method being peer reviewed

(d) subject matter experts able to judge the technical adequacy of non-PRA engineering evaluations important to the Level 2 PRA and to confirm that the applicable envelope defining the limits of the method are identified

(e) prohibited from reviewing work performed by a direct supervisor or work they have directly supervised

1-6.2.3 Specific Review Team Qualifications

The peer reviewer shall also be knowledgeable (by direct experience) of the specific method, code, tool, or approach (e.g., large event tree linking approach, MAAP [1-5] or MELCOR code [1-4], Technique for Human Error Rate Prediction (THERP) method [1-6], fragility assessment for equipment subject to severe accident environments) that was used in the PRA technical element assigned for review. Understanding and competence in the assigned area shall be demonstrated by the range of the individual's experience in the number of different, independent activities performed in the assigned area, as well as the different levels of complexity of these activities.

(a) One member of the peer-review team (the technical integrator) shall be familiar with all the PRA technical elements under review and shall have demonstrated the capability to integrate these PRA technical elements.

(b) The peer-review team shall have a team leader to lead the team in the performance of the review. The team leader need not be the technical integrator.

(c) The peer review shall have at least two reviewers dedicated to each reviewed technical element to ensure that consensus can be reached on the technical adequacy of the PRA being reviewed and be conducted over a period of time adequate to ensure that reviewed technical element receives the attention necessary to assess the technical adequacy.

(d) Exceptions to the requirements of this paragraph may be taken based on the nature of the PRA model change. A single-person peer review shall be justified only when the review involves an upgrade of a single element and the reviewer has acceptable qualifications for the technologies involved in the upgrade. All such exceptions shall be documented in accordance with Section 1-6.6 of this Standard. Regardless of any such exceptions, the collective qualification of the review team shall be appropriate to the scope of the peer review.

(e) If the peer reviewer is reviewing an NDM, the reviewer shall be knowledgeable of the technical subject addressed by the NDM. Understanding and competence of the NDM shall be demonstrated by the range of the individual's experience in that technical subject. Subject matter experts should be included to judge the technical adequacy of non-PRA engineering evaluations and to confirm that the applicable envelope defining the limits of the method are identified.

1-6.3 REVIEW OF PRA TECHNICAL ELEMENTS TO CONFIRM THE METHODS USED

The peer-review team shall use the requirements of this Section. The peer-review team shall review the technical requirements of the hazard group to determine if the method and the implementation of the method for each PRA technical element meet the requirements of this Standard. Additional material for those elements may be reviewed depending on the results obtained. The judgment of the reviewer shall be used to determine the specific scope and depth of the review in each PRA technical element and the need for walkdowns.

The results of the Level 2 PRA, including models and assumptions, and the results of each PRA technical element shall be reviewed to determine their reasonableness given the design and operation of the plant (e.g., investigation of cutset or sequence combinations for reasonableness).

Any NDM included in the scope of the peer review is reviewed against the requirements of Section 1-7. It is noted that an NDM can be peer reviewed within the scope of a plant PRA (i.e., concurrently with its implementation in a plant PRA) or via a dedicated stand-alone peer review. If NDMs are peer-reviewed concurrently with the implementations of methods, all specific requirements for the NDMs peer review shall be met. If the implementation of the method is peer reviewed in a separate peer review, only the applicable requirements for the scope of the review need to be met.

Even if exceptions to the requirements of Section 16.2.3(c) occur, concerning the composition of the peer-review team or the duration of the review, all SRs relevant to the scope of the peer review of the PRA are to be reviewed.

The extent of a focused-scope peer review includes all SRs (e.g., not just those for which significant deficiencies were cited), within the HLRs containing SRs with significant deficiencies. New significant deficiencies may be issued even for SRs that did not have previous significant deficiencies, as a focused-scope peer review encompasses all the SRs within an affected HLR.

1-6.4 EXPERT JUDGMENT

The use of expert judgment to implement requirements in this Standard shall be reviewed using the general requirements in Section 1-4.2.

1-6.5 PRA CONFIGURATION CONTROL

The peer-review team shall review the process, including implementation, for maintaining or upgrading the

PRA against the configuration control requirements of this Standard. The PRA configuration control program is reviewed against the requirements presented in Section 1-5.

1-6.6 DOCUMENTATION

1-6.6.1 Peer Review Team Documentation

The peer-review team's documentation shall demonstrate that the review process appropriately implemented the review requirements. Specifically, the peer-review documentation shall include the following:

(a) identification of the version of the PRA reviewed

(b) a statement of the scope of the peer review

(c) the names of the peer-review team members

(d) a brief resume for each team member describing the individual's employer, education, PRA training, and PRA and PRA technical element experience and expertise

(e) the elements of the PRA reviewed by each team member

(f) a discussion of the extent to which each PRA technical element was reviewed, including justification for any supporting requirements within the peer-review scope that were not reviewed

(g) results of the review identifying any differences between the requirements in the Technical Requirements section of Part 2 of this Standard and Section 1-5 and the method implemented, defined to a sufficient level of detail that will allow the resolution of the differences

(h) identification and significance of exceptions and gaps relative to this Standard's requirements, in sufficient detail to allow the resolution of the gaps that the peer reviewers have determined to be material to the PRA

(i) an assessment of PRA assumptions that the peer reviewers have determined to be material to the PRACE

(j) differences or dissenting views among peer reviewers

(k) recommended alternatives for resolution of any differences

(l) an assessment of the Capability Category of the SRs (i.e., identification of what Capability Category is met for the SRs)

(m) peer review consistent with newly developed methods requirements

1-6.6.2 Resolution of Peer Review Team Comments

Resolution of deficiencies against the requirements of this Standard that are identified by the peer-review team shall be documented. The resolution of these deficiencies shall describe how each was addressed such that the associated SR can be demonstrated to be met. The documentation shall indicate whether the deficiency is resolved via PRA maintenance or a PRA upgrade. The determination of whether the resolution adequately eliminates the deficiency shall be made by one or more individuals who meet the qualification requirements of Section 1-6.2.2.

Section 1-7
Newly Developed Methods

1-7.1 INTRODUCTION

This Section states requirements for Newly Developed Methods (NM) explicitly developed for use in PRA to support risk-informed decisions for nuclear power plants. The HLRs and SRs for the newly developed methods are contained in Table 1-7.2-1, Table 1-7.2-2, Table 1-7.2-3, Table 1-7.2-4, Table 1-7.2-5, Table 1-7.2-6, and Table 1-7.2-7.

1-7.2 OBJECTIVE

The objectives of the newly developed methods requirements are to ensure that a newly developed method is technically adequate and

(a) has a clearly defined scope and limitations

(b) is based on sound engineering and relevant science

(c) has proper treatment of assumptions and uncertainties

(d) is based on appropriate and well-understood data

(e) produces results that are consistent with expectations

(f) is clearly documented in such a way that knowledgeable personnel can understand it without ambiguity and that there is enough documentation so that it can be peer reviewed

These objectives are intended to be applicable to a large spectrum of methods, although it is understood that not all the SRs could be applicable to all methods. In some cases, depending on the method scope and purpose, some of the SRs may not be applicable. In addition, the SRs are designed to be able to address a stand-alone method (i.e., independent from its implementation on a specific plant PRA). It is recognized that, in some circumstances, a method can be so plant or site specific (especially in the external hazard domain) that a full review of the method can be performed only within its implementation. In such cases, it is envisioned that some of the Newly Developed Methods SRs could be overlapping with Part-specific SRs (e.g., SRs in Part 2). In such cases, the technical SRs in the appropriate Part may take priority over some Newly Developed Methods SRs.

Table 1-7.2-1 High Level Requirements for Newly Developed Methods (NM)

Designator	Requirement
HLR-NM-A	The purpose and scope of the newly developed method shall be clearly stated.
HLR-NM-B	The newly developed method shall be based on sound engineering and science relevant to its purpose and scope.
HLR-NM-C	The data (note that data can be numeric or non-numeric in nature) shall be relevant to the newly developed method, technically sound, and properly analyzed and applied.
HLR-NM-D	Uncertainties in the newly developed method shall be characterized. Sources of model uncertainties and related assumptions shall be identified.
HLR-NM-E	The results of the newly developed method shall be reproducible, reasonable, and consistent with the assumptions and data, given the purpose and scope of the newly developed method.
HLR-NM-F	The documentation of the newly developed method shall provide traceability of the work and facilitate incorporation of the newly developed method in a PRA model.

Table 1-7.2-2 Supporting Requirements for HLR-NM-A

The purpose and scope of the newly developed method shall be clearly stated (HLR-NM-A).

Index No. NM-A	Requirements
NM-A1	ENSURE that the stated purpose of the newly developed method (i.e., what is being achieved by the newly developed method) is consistent with the scope (established boundary) of the newly developed method.
NM-A2	ENSURE that the applicability and limitations of the newly developed method are consistent with the purpose and scope in SR NM-A1.
NM-A3	Based on the limitations and applicability of the newly developed method, IDENTIFY the areas of the PRA for which the newly developed method is intended to be used and those for which it is specifically not intended (e.g., hazards, technical elements, plant features, SRs impacted by the newly developed method).

Table 1-7.2-3 Supporting Requirements for HLR-NM-B

The newly developed method shall be based on sound engineering and science relevant to its purpose and scope (HLR-NM-B).

Index No. NM-B	Requirements
NM-B1	ESTABLISH the technical bases for the newly developed method by using approaches founded on established mathematical, engineering, and/or scientific principles (e.g., established through operating experience, tests, benchmarking, or acceptance by the scientific community).
NM-B2	If empirical models are used, ENSURE that they are supported by sufficient data, which are relevant to the newly developed method and, to the extent possible, that the experimental data have been shown to be repeatable.
NM-B3	IDENTIFY assumptions used to develop the technical bases of the newly developed method.
NM-B4	JUSTIFY the rationale for the assumptions identified in SR NM-B3 (e.g., backed by appropriate operational experience).

Table 1-7.2-4 Supporting Requirements for HLR-NM-C

The data (note that data can be numeric or non-numeric in nature) shall be relevant to the newly developed method, technically sound, and properly analyzed and applied (HLR-NM-C).

Index No. NM-C	Requirements
NM-C1	IDENTIFY the data needed in the development of the newly developed method (e.g., relevant plant-specific data, industry-wide current operating experience and data, or experimental or test data).
NM-C2	COLLECT relevant data consistent with current technical state of practice.
NM-C3	DEMONSTRATE that the data used, including experimental data or test data, are relevant to and support the technical basis of the newly developed method.
NM-C4	SPECIFY the basis for exclusion of data identified in SR NM-C1.
NM-C5	ANALYZE data (e.g., modifications to the data, use of data in a different context or beyond the original ranges, statistical analysis) using technically sound basis or criteria.
NM-C6	ENSURE that data are applied consistently with the purpose and scope of the newly developed method.

Table 1-7.2-5 Supporting Requirements for HLR-NM-D

Uncertainties in the newly developed method shall be characterized and their potential impact on the newly developed method understood (HLR-NM-D).

Index No. NM-D	Requirements
NM-D1 [Note (1)]	CHARACTERIZE the parameter uncertainties associated with the newly developed method consistent with the intended scope and purpose of the method; this characterization may include, for example, specifying the uncertainty range, qualitatively discussing the uncertainty range, or identifying the parameter estimate as conservative or bounding.
NM-D2	IDENTIFY the sources of model uncertainty associated with assumptions identified in SR NM-B3.
NM-D3	CHARACTERIZE the model uncertainties (identified in SR NM-D2) associated with the newly developed method; this characterization may be in the form of sensitivity studies.

NOTE:
(1) Depending on the purpose and scope of the method, uncertainty distributions may need to be explicitly calculated to allow for application of a method for risk-significant items to meet CC-II of related technical SRs in other Parts of this Standard.

Table 1-7.2-6 Supporting Requirements for HLR-NM-E

The results of the newly developed method shall be reproducible, reasonable, and consistent with the assumptions and data, given the purpose and scope of the newly developed method (HLR-NM-E).

Index No. NM-E	Requirements
NM-E1	REVIEW the results from the newly developed method to determine that they are reproducible, reasonable, and consistent with assumptions and data addressed in the SRs under HLR-NM-B and HLR-NM-C.
NM-E2	COMPARE the results of the newly developed method with existing methods and, when possible, IDENTIFY causes for substantial differences.
NM-E3	ENSURE uncertainties do not preclude meaningful use of the newly developed method results.

Table 1-7.2-7 Supporting Requirements for HLR-NM-F

The documentation of the newly developed method shall provide traceability of the work and facilitate incorporation of the newly developed method in a PRA model (HLR-NM-F).

Index No. NM-F	Requirements
NM-F1	DOCUMENT the newly developed method specifying what is used as input, the technical basis, and the implementation limitations by addressing the following, as well as other details needed to fully document how the set of the newly developed method SRs are satisfied: *(a)* the purpose and scope of the newly developed method *(b)* the intended use of the newly developed method *(c)* the limitations of the newly developed method *(d)* the technical basis for the newly developed method *(e)* the sources of data, the collection process and how the data is utilized to support of the newly developed method *(f)* the assumptions and uncertainties associated with the newly developed method *(g)* the interpretation of the results of the newly developed method in the framework of the intended use and application
NM-F2	DOCUMENT the intended process by which the newly developed method can be applied to a PRA model consistently with the intended use of the newly developed method and taking into account the purpose, scope, and limitations.

Section 1-8
References

References are cited here and in other Parts of this Standard as guides to the user. The user is cautioned that *(a)* the reference is not to be interpreted that there is a consensus approval on the technical acceptability of the reference and *(b)* there may be more recent versions of the references or alternative documents more pertinent to particular PRA applications.

[1-1] ASME/ANS RA-S-1.1-2024. Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications. The American Society of Mechanical Engineers and the American Nuclear Society.

[1-2] ANS/ASME-58.22-2014. Requirements for Low Power and Shutdown Probabilistic Risk Assessment. The American Nuclear Society and the American Society of Mechanical Engineers (Trial Use).

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

[1-4] SAND2017-0455 O (2017). MELCOR Computer Code Manuals. Sandia National Laboratories.

[1-5] EPRI TR-3002005285 (2015). Modular Accident Analysis Program 5 (MAAP5) Applications Guidance: Desktop Reference for Using Software." Electric Power Research Institute.

[1-6] NUREG/CR-1278 (1983). Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications. U.S. Nuclear Regulatory Commission.

[1-7] NUREG-1465 (1995). Accident Source Terms for Light Water Nuclear Power Plants. U.S. Nuclear Regulatory Commission.

[1-8] EPRI TR-105396 (1995). PSA Applications Guide. Electric Power Research Institute.

[1-9] Regulatory Guide 1.174, Rev. 3 (2018). An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis. U.S. Nuclear Regulatory Commission.

[1-10] NUREG/CR-6372 (1997). Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts. U.S. Nuclear Regulatory Commission.

[1-11] NUREG/CR-1563 (1996). Branch Technical Position on the Use of Expert Elicitation in the High-Level Radioactive Waste Program. U.S. Nuclear Regulatory Commission.

NONMANDATORY APPENDIX 1-A
MEANINGS OF ACTION VERBS

This Standard uses action verbs to state requirements. Dictionaries provide multiple meanings for most verbs. Table 1-A-1 states, with examples, the meanings of action verbs as used in this Standard. The relevant dictionary meanings were derived from American dictionaries [e.g., Random House Unabridged (dictionary.com), Merriam-Webster (merriam-webster.com)] with a few modifications to address specific usage in this Standard.

Table 1-A-1 List of Action Verbs

Action Verb	Relevant Dictionary Meaning	Examples of Usage in This Standard
ADDRESS	To direct the efforts or attention to	An example of the appropriate usage of the action verb can be found in SR CP-B4 in Part 2, which states ADDRESS the impact of material deterioration due to aging and in-service degradation of structures for the material composition and material properties determined in SR CP B1.
ANALYZE	To examine critically so as to bring out the essential elements	An example of the appropriate usage of the action verb can be found in SR NM-C5 in Part 1, which states ANALYZE data (e.g., modifications to the data, use of data in a different context or beyond the original ranges, statistical analysis) using technically sound basis or criteria.
ASSESS	To determine the importance, size, or value of	An example of the appropriate usage of the action verb can be found in SR PT-F4 in Part 2, which states ASSESS the effects of model uncertainties.
ASSUME	To take for granted without proof	An example of the appropriate usage of this action verb can be found in SR PT-B8 in Part 2, where CC-I states ASSUME that secondary containment or auxiliary building(s) do not act as an effective radionuclide barrier to radionuclide release for containment failure sequences with core damage.
CALCULATE	To determine by mathematical processes, compute	CALCULATE involves a mathematical process, whereas ESTIMATE does not necessarily involve a calculation (e.g., quantification of a probability or frequency) and can be derived qualitatively. An example of the appropriate usage of the action verb can be found in SR CP-D2 in Part 2, where CC-II states CALCULATE the uncertainty distribution for the containment failure criteria in the form of a fragility curve.
CHARACTERIZE	To describe the character or quality of	In this Standard, CHARACTERIZE is used with respect to sources of uncertainty. An example of the appropriate usage of the action verb can be found in SR CP-D2 in Part 2, where CC-I states CHARACTERIZE the uncertainty interval of the threshold for containment failure by specifying or discussing the range of the uncertainty, consistent with the characterization of parameter uncertainties.
COLLECT	To bring together into one body or place	An example of the appropriate usage of the action verb can be found in SR NM-C2 in Part 1, which states COLLECT relevant data consistent with current technical state of practice.
COMPARE	To examine the character or qualities of especially in order to discover similarities or differences	An example of the appropriate usage of the action verb can be found in SR PT-E14 in Part 2, which states COMPARE results to those from similar plants if information from similar plants is available.
DEFINE	To determine or identify the essential qualities or meaning of	An example of the appropriate usage of the action verb can be found in SR SA-A1 in Part 2, which states DEFINE the objectives of deterministic analysis performed to support the Level 2 PRA.
DEMONSTRATE	To prove or make clear by reasoning or evidence	An example of the appropriate usage of the action verb can be found in SR ST-A6 in Part 2, which states DEMONSTRATE that combinations of attributes defined in SR ST-A5 lead to a complete set of RCs.

Table 1-A-1 List of Action Verbs (Cont'd)

Action Verb	Relevant Dictionary Meaning	Examples of Usage in This Standard
DETERMINE	To find out or come to a decision about by investigation, reasoning, or calculation	An example of the appropriate usage of the action verb can be found in SR CP-B1 in Part 2, which states DETERMINE the "as built" structural geometry, material composition and material properties.
DEVELOP	To bring out the capabilities or possibilities of	An example of the appropriate usage of the action verb can be found in SR PT-A2 in Part 2, which states DEVELOP a logic structure using the method selected in SR PT-A1 (i.e., severe accident progression method, CET or equivalent).
DOCUMENT	To furnish documentary evidence of	An example of the appropriate usage of the action verb can be found in SR L1-C2 in Part 2, which states DOCUMENT sources of model uncertainty, the related assumptions, their characterization and reasonable alternatives associated with the Level 1/Level 2 PRA Interface.
ENSURE	To make sure or certain	An example of the appropriate usage of the action verb can be found in SR SA-D3 in Part 2, which states ENSURE the reasonableness and acceptability of the calculated results defined in SR SA-A2.
ESTABLISH	To bring into being on a firm basis	An example of the appropriate usage of the action verb can be found in SR NM-B1 in Part 1, which states ESTABLISH the technical bases for the newly developed method.
ESTIMATE	To form an approximate judgment or opinion regarding the value, amount, size, and so on; to calculate approximately	ESTIMATE does not necessarily involve a calculation (e.g., quantification of a probability or frequency), and an estimate can be derived qualitatively, whereas CALCULATE involves a mathematical process. An example of the appropriate usage of the action verb can be found in SR CP-C1 in Part 2, where CC-I states ESTIMATE conservative containment overpressure failure probabilities as a function of discrete combinations of independent variables (e.g., a fragility curve at various temperatures and pressures) using the method selected in SR CP-B3.
EVALUATE	To determine or set the value or amount of; appraise	An example of the appropriate usage of the action verb can be found in SR PT-E2 in Part 2, where CC-I states EVALUATE dependencies introduced by common physical parameters involved in multiple CET top events (or equivalent) in a conservative or a combination of conservative and realistic manner.
EXPLAIN	To make plain, clear, or intelligible	An example of the appropriate usage of the action verb can be found in SR PT-E5 in Part 2, which states EXPLAIN any discrepancies between the sum of the CDF contributors and the total Level 2 end state frequencies.
IDENTIFY	To recognize or establish as being a particular thing	An example of the appropriate usage of the action verb can be found in SR L1-B9 in Part 2, which states IDENTIFY sources of uncertainty, the related assumptions, and reasonable alternatives of the Level 1/Level 2 PRA Interface.
IMPLEMENT	To put into effect according to or by means of a definite plan or procedure	An example of the appropriate usage of the action verb can be found in SR CC-A1 in Part 1, which states IMPLEMENT a process to track changes, PRA technology, and so on.
INCLUDE	To place in an aggregate, class, category, or the like	An example of the appropriate usage of the action verb can be found in SR CP-A2 in Part 2, which states INCLUDE, as applicable, containment failure mechanisms resulting from a list of items.
JUSTIFY	To show a satisfactory reason for some action	An example of the appropriate usage of the action verb can be found in SR L1-B7 in Part 2, where CC-I states JUSTIFY the criteria used (qualitative or quantitative or combination) to ensure that a sufficient number of accident sequences are being transferred.
MODEL	To create a representation of	An example of the appropriate usage of the action verb can be found in SR PT-A11 in Part 2, which states MODEL the logical dependencies between systems, components and human actions in the Level 1 PRA and in the logic structure developed in SR PT-A2.
PERFORM	To carry out; execute; do	An example of the appropriate usage of the action verb can be found in SR PT-C7 in Part 2, where CC-II states PERFORM a realistic secondary side isolation capability analysis for significant accident progression sequences caused by SGTR (if applicable).

Table 1-A-1 List of Action Verbs (Cont'd)

Action Verb	Relevant Dictionary Meaning	Examples of Usage in This Standard
PROVIDE	To furnish, supply or equip	An example of the appropriate usage of the action verb can be found in SR SA-E2 in Part 2, which in CC-II states PROVIDE variations in input parameters particular to the modeling tool selected in SR SA-C1 that impact the uncertainty of the models or assumptions defined in SR SA-E1.
REVIEW	To go over or examine critically or deliberately	An example of the appropriate usage of the action verb can be found in SR PT-E10 in Part 2, which states REVIEW the importance measures evaluated in SR PT-E9 to ensure that they are consistent with expected results and understand or reconcile the reason for any unexpected results.
SATISFY	To give assurance to; to answer sufficiently	The use of SATISFY is exclusively directed to fulfilling requirements stipulated elsewhere in this Standard. An example of the appropriate usage of the action verb can be found in SR SA-B1 in Part 2, which states if using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment.
SELECT	To choose in preference to another or others; pick out	An example of the appropriate usage of the action verb can be found in SR CP-B3 in Part 2, CC-I which states SELECT a methodology to determine the fragility curve of the containment pressure boundary for both rupture and excess leakage failure modes identified in SR CP-A8.
SPECIFY	To name or state explicitly or in detail	An example of the appropriate usage of the action verb can be found in SR CP-C2 in Part 2, where CC-I states SPECIFY conservative containment failure probabilities for each containment failure mechanism identified in SR CP-A8 and for each plant operational mode included in the Level 2 analysis that are not associated with SR CP-C1 using the methods selected in SR CP-B7 and SR CP-B8.
TRANSFER	To convey from one person, place, or situation to another; move or shift	An example of the appropriate usage of the action verb can be found in SR L1-B7 in Part 2, where CC-II states TRANSFER all accident sequences that are included in CDF from the Level 1 PRA to the Level 2 PRA including the Level 1 uncertainty distributions.
USE	To employ for some purpose, make use of	An example of the appropriate usage of the action verb can be found in SR PT-C5 in Part 2, CC-I which states USE conservative or a combination of conservative and realistic treatments of adverse environmental impacts for assessing equipment survivability for equipment inside containment.

PART 2 TECHNICAL REQUIREMENTS FOR LEVEL 2 PRA

Section 2-1 Overview of Level 2 PRA Requirements

2-1.1 PRA SCOPE

This section provides requirements for each of the technical elements that compose the Level 2 analysis. The scope of a Level 2 analysis covered by this Standard includes the determination of the progression of severe accidents from core damage through radionuclide release to the environment or the determination that a release will not occur. The scope of this Standard addresses postulated accident sequences initiated from all modes of reactor operation (at-power, shutdown, and transition states) and by internal events, internal hazards, and/or external hazards addressed in ASME/ANS RA-S-1.1-2024 [2-1]. As a result, it is expected that the requirements described here are applied separately to Level 1 PRA results for each plant operating state (POS), as addressed in ASME/ANS-58.22-2014 (Trial Use) [2-2].

The requirements address the analysis of the various phenomena that occur inside the reactor vessel, the containment structure, and possibly other structures involved in the fission product release pathway. The results of the Level 2 analysis may be the final endpoint of the probabilistic analysis or may be used as input to a Level 3 analysis (i.e., consequence analysis).

The requirements of this section, which are organized by six technical elements that compose the analysis considered necessary to extend the Level 1 PRA from core damage to radionuclide release categories, are as follows:

(a) Level 1/Level 2 PRA Interface (L1)
(b) Containment Performance Analysis (CP)
(c) Severe Accident Progression Analysis (SA)
(d) Probabilistic Treatment of Accident Progression and Source Terms (PT)
(e) Source Term Analysis (ST)
(f) Evaluation and Presentation of Results (ER)

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

Section 2-2
Level 2 PRA Technical Elements and Requirements

2-2.1 LEVEL 1/LEVEL 2 PRA INTERFACE (L1)

2-2.1.1 Objectives

The objectives of the Level 1/Level 2 PRA Interface are to provide an effective transfer of information between the Level 1 PRA evaluation of CDF and the core melt progression analysis that is treated in the Level 2 analysis in such a way that:

(a) The interface boundary between the Level 1 analysis and the Level 2 analysis is defined in a manner that preserves the transfer of pertinent information (e.g., dependencies) from the Level 1 PRA to the Level 2 PRA.

(b) The methodology is clear, is consistent with the Level 1 PRA evaluation, and it creates an adequate transition from the Level 1 PRA.

(c) The Level 1/Level 2 PRA Interface is documented to provide traceability of the work.

Table 2-2.1-1 provides the HLRs for the Level 1/Level 2 PRA Interface.

Table 2-2.1-1 High Level Requirements for the Level 1/Level 2 PRA Interface (L1)

Designator	Requirement
HLR-L1-A	A method shall be specified to ensure that information required for the Level 2 PRA is transferred from the Level 1 PRA and supplemented as needed to support the Level 2 PRA accident progression analysis.
HLR-L1-B	A method shall be implemented to transfer necessary information (e.g., accident sequences and corresponding frequencies, dependencies, and system successes) from the Level 1 PRA analysis to the Level 2 PRA.
HLR-L1-C	The documentation of the Level 1/Level 2 PRA Interface shall provide traceability of the work.

Table 2-2.1-2 Supporting Requirements for HLR-L1-A

A method shall be specified to ensure that information required for the Level 2 PRA is transferred from the Level 1 PRA and supplemented as needed to support the Level 2 PRA accident progression analysis (HLR-L1-A).

Index No. L1-A	Capability Category I	Capability Category II
L1-A1	IDENTIFY the physical characteristics at the time of core damage that can influence the major features of severe accident progression, containment performance, and radionuclide release that are necessary to effectively transfer information to the Level 2 analysis. The following physical characteristics should be included as applicable: *(a)* RCS status (e.g., pressure and configuration for modeled plant operating states) *(b)* status of emergency core coolant systems (failure in injection can result in a dry cavity and extensive core concrete interaction) *(c)* status of containment heat removal systems, hydrogen mitigation systems, and venting systems *(d)* containment integrity (e.g., open, intact, vented, bypassed, or failed) and steam generator tube integrity [pressurized water reactors (PWRs)] *(e)* status of containment inerting [boiling water reactors (BWRs)] *(f)* status of support systems, nonsafety systems, and portable equipment *(g)* time of core damage after the initiating event *(h)* environmental or physical conditions introduced by the initiating event or hazard, if any, that may interfere with recovery actions that would occur after the onset of core damage *(i)* initial state of fuel in the reactor *(j)* design and physical configuration of primary coolant system, primary and secondary containment, and other neighboring structures, if included within the scope of the analysis *(k)* plant-specific or other physical characteristics important for capturing major features of severe accident progression, containment performance, or radionuclide release	
L1-A2	IDENTIFY the accident sequence characteristics that impact physical characteristics identified in SR L1-A1.	
L1-A3	IDENTIFY where the physical characteristics identified in SR L1-A1 and the accident sequence characteristics identified in SR L1-A2 are specified in the probabilistic logic model(s) (i.e., Level 1 PRA, bridge tree, or Level 2 PRA).	
L1-A4	JUSTIFY any characteristics identified in SR L1-A1 or SR L1-A2 that are excluded from the severe accident progression, containment performance, and radionuclide release categories analysis (e.g., due to plant design or operational considerations).	
L1-A5	Using the characteristics identified in SR L1-A1 and SR L1-A2, SPECIFY a method for transferring necessary input information from the Level 1 PRA accident sequences and any supplemental analyses to the Level 2 PRA.	

Table 2-2.1-3 Supporting Requirements for HLR-L1-B

A method to transfer necessary information (e.g., accident sequences and corresponding frequencies, dependencies and system successes) from the Level 1 PRA analysis to the Level 2 PRA shall be implemented (HLR-L1-B).

Index No. L1-B	Capability Category I	Capability Category II
L1-B1	ENSURE that the Level 2 PRA is based on a documented, peer-reviewed Level 1 PRA with adequate scope to support the Level 2 PRA, and significant deficiencies identified during the peer review for the Level 1 PRA that are relevant to the Level 2 PRA are resolved and incorporated into the development of the Level 2 PRA.	
L1-B2	IDENTIFY the dependencies to be included in transferring information from Level 1 PRA to Level 2 PRA logic models.	
L1-B3	INCLUDE dependencies between the Level 1 PRA and Level 2 PRA models as identified in SR L1-B2 in implementing the method specified in SR L1-A5.	
L1-B4	INCLUDE system successes in addition to system failures in the evaluation of accident sequences to the extent needed for estimation of RC frequencies.	

Table 2-2.1-3 Supporting Requirements for HLR-L1-B (Cont'd)

Index No. L1-B	Capability Category I	Capability Category II
L1-B5	SPECIFY sufficient accident sequence end states to support bounding estimates of radionuclide release categories.	SPECIFY sufficient accident sequence end states to support realistic estimates of radionuclide RCs for significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
L1-B6	If accident sequences are grouped, ENSURE that the transfer of information from Level 1 PRA to Level 2 PRA provides a conservative representation of accident progression sequences in the Level 2 PRA logic model.	If accident sequences are grouped, ENSURE that the transfer of information from Level 1 PRA to Level 2 PRA provides a realistic representation of significant accident progression sequences in the Level 2 PRA logic model. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
L1-B7	Using the method specified in SR L1-A5, TRANSFER accident sequences that are included in CDF from the Level 1 PRA into the Level 2 PRA such that CDF is preserved and premature truncation of accident sequences that may be important in the characterization of the radionuclide release or sequences that defeat all or most containment mitigation measures does not occur. JUSTIFY the criteria used (qualitative or quantitative or combination) to ensure that a sufficient number of accident sequences are being transferred (e.g., neglected accident sequences are not expected to impact the scope of the Level 2 PRA surrogate release metric).	Using the method specified in SR L1-A5, TRANSFER all accident sequences that are included in CDF from the Level 1 PRA to the Level 2 PRA including the Level 1 uncertainty distributions.
L1-B8	TRANSFER Level 1 model uncertainties and assumptions for inclusion in the Level 2 analysis.	
L1-B9	IDENTIFY the sources of uncertainty, the related assumptions, and reasonable alternatives of the Level 1/Level 2 PRA Interface in a manner that supports the applicable requirements of SR PT-F4.	
L1-B10	For the sources of parameter uncertainty identified in SR L1-B9, CHARACTERIZE the uncertainty range for the parameters.	For the sources of parameter uncertainty identified in SR L1-B9, CALCULATE a mean value of the parameters used and PROVIDE a statistical representation of the uncertainty intervals for the parameter estimates of significant basic events. Expert judgment may be used to estimate the uncertainty range for parameters where alternate approaches are impractical. If using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment. For basic events that are not significant, ENSURE the requirement for CC-I is met.

Table 2-2.1-4 Supporting Requirements for HLR-L1-C

The documentation of the Level 1/Level 2 PRA Interface shall provide traceability of the work (HLR-L1-C).

Index No. L1-C	Capability Category I	Capability Category II
L1-C1	DOCUMENT the process used in the Level 1/Level 2 PRA Interface specifying what is used as input, the applied methods and results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* The method used to ensure information required for the Level 2 PRA is transferred from the Level 1 PRA and is supplemented as needed to support the Level 2 PRA accident progression analysis. *(b)* Level 1 PRA attributes that are included in the Level 1/Level 2 PRA Interface. *(c)* The methods and criteria used to propagate information across the Level 1/Level 2 PRA Interface and address dependencies.	
L1-C2	DOCUMENT sources of model uncertainty, related assumptions, their characterization, and reasonable alternatives associated with the Level 1/Level 2 PRA Interface identified in SR L1-B9 and SR L1-B10.	

2-2.2 CONTAINMENT PERFORMANCE ANALYSIS (CP)

2-2.2.1 Objectives

The objectives of the Containment Performance Analysis are to analyze the capacity of the containment structure in such a way that:

(a) The mechanisms of containment failure are identified.

(b) Methods to develop containment failure probabilities for containment failure mechanisms are selected.

(c) The capacity of the containment to withstand loads from core damage accidents is determined.

(d) Uncertainties in containment failure analysis are identified and characterized.

(e) The Containment Performance Analysis is documented to provide traceability of the work.

Table 2-2.2-1 provides the HLRs for the Containment Performance Analysis.

Table 2-2.2-1 High Level Requirements for Containment Performance Analysis (CP)

Designator	Requirement
HLR-CP-A	The mechanisms of containment failure shall be identified as input to the assessment of severe accident containment integrity.
HLR-CP-B	Methods shall be selected to develop containment failure probabilities for the failure mechanisms identified in HLR-CP-A.
HLR-CP-C	Containment overpressure fragility curves and failure probabilities for other severe accident induced containment boundary challenges shall be determined.
HLR-CP-D	Uncertainties in the Containment Performance Analysis shall be characterized. Sources of model uncertainty and related assumptions shall be identified and their potential impact on the analysis understood.
HLR-CP-E	The documentation of the Containment Performance Analysis shall provide traceability of the work.

Table 2-2.2-2 Supporting Requirements for HLR-CP-A

The mechanisms of containment failure shall be identified as input to the assessment of severe accident containment integrity (HLR-CP-A).

Index No. CP-A	Capability Category I	Capability Category II
CP-A1	IDENTIFY potential containment failure mechanisms including consideration of severe accident analyses for similar plant designs, plant-specific severe accident challenges, and unique design features.	
CP-A2	IDENTIFY potential containment failure mechanisms caused by severe accident phenomena that directly challenge containment integrity. INCLUDE, as applicable, containment failure mechanisms resulting from *(a)* gradual containment overpressure *(b)* containment overpressure from hydrogen combustion (deflagration and detonation) *(c)* containment overpressure from direct containment heating phenomena *(d)* containment penetration seal failure/degradation associated with sustained high temperature exposure *(e)* hydrodynamic loads generated from high-pressure blowdown of steam and non-condensable gases into the suppression pool (or equivalent) *(f)* steam explosions within large open or enclosed water pools in contact with containment boundary *(g)* direct contact of corium debris with containment boundary materials/surfaces *(h)* radiation damage to containment penetrations *(i)* core-concrete attack leading to basemat melt-through (BMMT)	
CP-A3	In identifying the failure mechanisms for the containment structure in SR CP-A2, INCLUDE failure mechanisms associated with concrete cracking, liner tearing, failure or closure status of doors, hatches, mechanical penetrations, electrical assemblies, and bellows seals as applicable to the plant and POS being evaluated in the containment failure analysis.	
CP-A4	IDENTIFY potential containment failure mechanisms caused by severe accident phenomena that can indirectly challenge containment integrity. INCLUDE, as applicable, containment failure mechanisms resulting from *(a)* erosion, displacement, or over-stressing of structures internal to the containment due to severe accident phenomena causing a potential loss of containment integrity *(b)* energetic failure of the reactor vessel at high pressure resulting from "missile" impact with containment boundary *(c)* thermo-chemical erosion of a concrete reactor pedestal that might result in displacement of the reactor pressure vessel *(d)* movement of appended piping and structural damage to piping penetrations passing through the containment pressure boundary	
CP-A5	For core damage accident sequences initiated by external hazards, IDENTIFY containment failure mechanisms and degraded conditions caused by the evaluated external hazards in the containment failure analysis for modeled POSs.	
CP-A6	IDENTIFY pre-existing failure modes or plant conditions that compromise containment capability to withstand severe accident challenges.	
CP-A7	If buildings outside the containment pressure boundary are assumed to participate in the release pathway for fission products released to the environment, IDENTIFY potential failure mechanisms that could compromise the ability of these buildings to retain fission products as a consequence of severe accident progression.	
CP-A8	IDENTIFY those failure mechanisms from SR CP-A1, SR CP-A2, SR CP-A3, SR CP-A4, SR CP-A5, SR CP-A6, and SR CP-A7 that are to be addressed in the assessment of containment performance in HLR-CP-B.	
CP-A9	JUSTIFY exclusion of those containment failure mechanisms that are identified in SR CP-A1, SR CP-A2, SR CP-A3, SR CP-A4, SR CP-A5, SR CP-A6, and SR CP-A7 that are not included in the list generated in SR CP-A8 (e.g., exclusion based on nonapplicability to plant design, POS, or containment design).	

Table 2-2.2-3 Supporting Requirements for HLR-CP-B

Methods shall be selected to develop containment failure probabilities for the failure mechanisms identified in HLR-CP-A (HLR-CP-B).

Index No. CP-B	Capability Category I	Capability Category II
CP-B1	DETERMINE the "as built" structural geometry, material composition, and material properties to be used in the Containment Performance Analysis.	
CP-B2	DETERMINE failure properties, including capacity limit(s), as appropriate, for structural materials included in the containment integrity analysis.	
CP-B3	SELECT a methodology to determine the fragility curve of the containment pressure boundary for both rupture and excess leakage failure modes identified in SR CP-A8. JUSTIFY the methodology as resulting in a conservative assessment of the containment ultimate strength and excess leakage failure modes (e.g., no credit for plastic deformation). If generic assessments for similar plants are used, JUSTIFY applicability to the plant being evaluated (e.g., similar containment designs or estimating containment capacity based on design pressure and a conservative multiplier relating containment design pressure and median ultimate failure pressure). INCLUDE consideration of the potential for detonations where quasi-static containment capability evaluations may not be adequate.	SELECT a methodology to determine the fragility curve of the containment pressure boundary for both rupture and excess leakage failure modes identified in SR CP-A8. The method is to rely on a validated computational model that evaluates structural response based on mathematical expressions and correlations that reflect material behavior and governing physical processes and is applicable to the plant-specific configuration and conditions for the significant accident progression sequences. INCLUDE consideration of the potential for detonations where quasi-static containment capability evaluations may not be adequate.
CP-B4	ADDRESS the impact of material deterioration due to aging and in-service degradation of structures for the material composition and material properties determined in SR CP-B1.	
CP-B5	SPECIFY bounding quasi-static thermal-mechanical properties or the physical attributes of challenges on the containment structure used to evaluate the containment failure mechanisms identified in SR CP-A8.	SPECIFY plant-specific realistic quasi-static thermal-mechanical loads or the physical attributes of challenges on the containment structure used to evaluate the containment failure mechanisms identified in SR CP-A8 for significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
CP-B6	ASSUME buildings outside containment structures do not have the capacity to withstand thermal or mechanical loads generated during core damage accident sequences.	DETERMINE the fragility curve for enclosed structures outside the containment pressure boundary to survive loads generated by accident progression sequences in which the structure lies in the fission product release pathway to the environment.
CP-B7	For direct containment failure mechanisms identified in SR CP-A8 not addressed in SR CP-B3, SELECT a conservative method for assigning containment failure probabilities given the plant-specific spectrum of severe accident challenges.	For direct containment failure mechanisms that represent a significant containment challenge that are identified in SR CP-A8 but not addressed in SR CP-B3, SELECT a realistic method for assigning containment failure probabilities given the plant-specific spectrum of severe accident challenges. For containment challenges that are not significant, ENSURE the requirement for CC-I is met.
CP-B8	For indirect containment failure mechanisms identified in SR CP-A8, SELECT a conservative method for assigning containment failure probabilities given the plant-specific spectrum of severe-accident challenges.	For indirect containment failure mechanisms that represent a significant containment challenge that are identified in SR CP-A8, SELECT a realistic method for assigning containment failure probabilities given the plant-specific spectrum of severe-accident challenges. For containment challenges that are not significant, ENSURE the requirement for CC-I is met.

Table 2-2.2-4 Supporting Requirements for HLR-CP-C

Containment overpressure fragility curves and failure probabilities for other severe accident induced containment boundary challenges shall be determined (HLR-CP-C).

Index No. CP-C	Capability Category I	Capability Category II
CP-C1	ESTIMATE conservative containment overpressure failure probabilities as a function of discrete combinations of independent variables (e.g., a fragility curve at various temperatures and pressures) using the method selected in SR CP-B3. If using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment.	CALCULATE realistic containment overpressure failure probabilities as a function of discrete combinations of independent variables (e.g., one or more fragility curves at various temperatures) using the method selected in SR CP-B3 for significant containment challenges. For containment challenges that are not significant, ENSURE the requirement for CC-I is met. If using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment.
CP-C2	SPECIFY conservative containment failure probabilities for each containment failure mechanism identified in SR CP-A8 and for each plant operational mode included in the Level 2 analysis that are not associated with SR CP-C1 using the methods selected in SR CP-B7 and SR CP-B8.	SPECIFY realistic containment failure probabilities for each containment failure mechanism identified in SR CP-A8 and for each plant operational mode included in the Level 2 analysis that are not associated with SR CP-C1 using the methods selected in SR CP-B7 and SR CP-B8 for significant containment challenges. For containment challenges that are not significant, ENSURE the requirement for CC-I is met.
CP-C3	For each containment failure mechanism identified in SR CP-A8 and for each plant operational mode included in the Level 2 analysis, SPECIFY a conservative failure location and a conservative final opening size in the containment pressure boundary; JUSTIFY applicability of generic and other analyses used (e.g., similar failure locations in similar containment designs). If multiple (alternate) failure locations and/or opening sizes are considered to apply to a specific failure mechanism, ESTIMATE conditional probabilities assigned to each possibility for significant containment challenges.	For each containment failure mechanism identified in SR CP-A8 and for each plant operational mode included in the Level 2 analysis, SPECIFY the failure location and a realistic value of the final opening size in the containment pressure boundary as a function of pressure for significant containment challenges. If multiple (alternate) failure locations and/or opening sizes are considered to apply to a specific failure mechanism, ESTIMATE conditional probabilities assigned to each possibility for significant containment challenges. For containment challenges that are not significant, ENSURE the requirement for CC-I is met.
CP-C4	For external hazards included in the Level 2 PRA, conservatively ESTIMATE the impact of the hazard on the failure mechanisms identified in SR CP-A5. JUSTIFY the use of generic fragility data. For external hazard fragilities, ENSURE the evaluation meets at least the Capability Category I requirements in [2-1] as applicable to the scope of the Level 2 PRA or SPECIFY a basis to support the claim of nonapplicability of any of these requirements: *(a)* Part 5–Seismic Event PRA: HLR SFR-A through HLR SFR-F *(b)* Part 7–High Wind PRA: HLR-WFR-A through HLR-WFR-I *(c)* Part 8–External Flood PRA: HLR-XFFR-A through HLR-XFFR-F *(d)* Part 9–Other Hazard PRA: HLR-XFR-A through HLR-XFR-B If using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment.	For external hazards included in the Level 2 PRA, CALCULATE a realistic assessment of the hazard impact on the containment failure mechanisms identified in SR CP-A5. For external hazard fragilities, ENSURE the evaluation meets the Capability Category II requirements in [2-1] as applicable to the scope of the Level 2 PRA or SPECIFY a basis to support the claim of nonapplicability of any of these requirements: *(a)* Part 5–Seismic Event PRA: HLR SFR-A through HLR SFR-F *(b)* Part 7–High Wind PRA: HLR-WFR-A through HLR-WFR-I *(c)* Part 8–External Flood PRA: HLR-XFFR-A through HLR-XFFR-F *(d)* Part 9–Other Hazard PRA: HLR-XFR-A through HLR-XFR-B If using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment.
CP-C5	IDENTIFY method-specific limitations in the Containment Performance Analysis.	

Table 2-2.2-5 Supporting Requirements for HLR-CP-D

Uncertainties in the Containment Performance Analysis shall be characterized. Sources of model uncertainty and related assumptions shall be identified and their potential impact on the analysis understood (HLR-CP-D).

Index No. CP-D	Capability Category I	Capability Category II
CP-D1	IDENTIFY the sources of uncertainty, the related assumptions, and reasonable alternatives of the Containment Performance Analysis in a manner that supports the applicable requirements of SR PT-F4.	
CP-D2	CHARACTERIZE the uncertainty interval of the threshold for containment failure by specifying or discussing the range of the uncertainty, consistent with the characterization of parameter uncertainties.	CALCULATE the uncertainty distribution for the containment failure criteria in the form of a fragility curve.
CP-D3	CHARACTERIZE the uncertainty in the final opening size, location, and probability of the containment failure specified in SR CP-C3.	
CP-D4	CHARACTERIZE sources of parameter uncertainty, modeling uncertainty, and assumptions identified in SR CP-D1 (e.g., how the containment strength or resistance to failure is affected).	

Table 2-2.2-6 Supporting Requirements for HLR-CP-E

The documentation of the Containment Performance Analysis shall provide traceability of the work (HLR-CP-E).

Index No. CP-E	Capability Category I	Capability Category II
CP-E1	DOCUMENT the process used in the Containment Performance Analysis specifying what is used as input, the applied methods, and results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the mechanisms of containment failure *(b)* the methods used to develop containment failure probabilities *(c)* the containment overpressure fragility curves and failure probabilities for other severe accident induced containment boundary challenges *(d)* geometric configuration(s) of containment *(e)* material composition and material properties *(f)* type and extent of material or geometric degradation due to adverse environmental conditions *(g)* criteria to define what containment failure mechanisms were excluded	
CP-E2	DOCUMENT the containment failure criteria (thresholds or fragility curve) and technical rationale defined for each containment failure mechanism.	
CP-E3	DOCUMENT the containment failure probabilities defined for each containment failure mechanism and support each criterion with a technical justification.	
CP-E4	DOCUMENT the technical basis for the containment failure location and opening size (or leak rate) resulting from each containment failure mechanism and the technical basis for the associated probabilities.	
CP-E5	DOCUMENT the sources of model uncertainty, related assumptions, their characterization, and reasonable alternatives associated with the Containment Performance Analysis identified in SR CP-D1.	

2-2.3 SEVERE ACCIDENT PROGRESSION ANALYSIS (SA)

2-2.3.1 Objectives

The objectives of the Severe Accident Progression Analysis to support a Level 2 PRA are to evaluate the progression of events in as realistic a manner as practical and in a manner consistent with the degree of realism of the other attributes of the Level 2 PRA in such a way that

(a) the Severe Accident Progression Analysis defines objectives and quantitative parameters or metrics

(b) the assumptions supporting deterministic calculations are identified and appropriate input parameters are estimated

(c) appropriate Severe Accident Progression Analysis tool(s) are selected

(d) severe accident progression analyses to support the probabilistic accident progression framework are performed

(e) assumptions and uncertainties are identified and characterized

(f) the Severe Accident Progression Analysis is documented to provide traceability of the work

Table 2-2.3-1 provides the HLRs for the Severe Accident Progression Analysis.

Table 2-2.3-1 High Level Requirements for Severe Accident Progression Analysis (SA)

Designator	Requirement
HLR-SA-A	The objectives of severe accident progression analysis and the quantitative parameters that deterministic analysis will calculate shall be defined.
HLR-SA-B	Assumptions used to perform deterministic calculations shall be identified, and values of input parameters shall be estimated.
HLR-SA-C	An appropriate deterministic method shall be selected for generating the quantitative parameters defined in HLR-SA-A.
HLR-SA-D	Severe Accident Progression Analysis calculations shall be performed as needed to support the probabilistic accident progression framework.
HLR-SA-E	Uncertainties in the Severe Accident Progression Analysis shall be characterized. Sources of model uncertainty and related assumptions shall be identified and their potential impact on the analysis understood.
HLR-SA-F	The documentation of the Severe Accident Progression Analysis shall provide traceability of the work.

Table 2-2.3-2 Supporting Requirements for HLR-SA-A

The objectives of severe accident progression analysis and the quantitative parameters that deterministic analysis will calculate shall be defined (HLR-SA-A).

Index No. SA-A	Capability Category I	Capability Category II
SA-A1	DEFINE the objectives of severe accident progression analyses performed to support the Level 2 PRA.	
SA-A2	SPECIFY the output parameters to be calculated by the severe accident progression analyses performed to support the Level 2 PRA.	

Table 2-2.3-3 Supporting Requirements for HLR-SA-B

Assumptions used to perform deterministic calculations shall be identified, and values of input parameters shall be estimated (HLR-SA-B).

Index No. SA-B	Capability Category I	Capability Category II
SA-B1	ESTIMATE values of accident parameters for deterministic calculations using conservative methods and assumptions. If using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment.	ESTIMATE values of accident parameters for deterministic calculations using realistic methods and assumptions for significant accident progression sequences. If using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
SA-B2	JUSTIFY values of input parameters applied in deterministic calculations (e.g., values are based on consensus method or values are realistic for the specific plant).	

Table 2-2.3-4 Supporting Requirements for HLR-SA-C

An appropriate deterministic method shall be selected for generating the quantitative parameters defined in HLR-SA-A (HLR-SA-C).

Index No. SA-C	Capability Category I	Capability Category II
SA-C1	SELECT a deterministic method for generating the output parameters specified in SR SA-A2. USE a reference plant calculation or a calculation derived from first principles and/or well-established correlations using conservative, or a combination of conservative and realistic methods that provide conservative results, in aggregate.	SELECT a deterministic method for generating the output parameters specified in SR SA-A2. USE a realistic modeling tool that reflects plant-specific design features and knowledge of severe accident behavior that is validated against available experimental data or other established benchmarks for significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
SA-C2	SPECIFY a basis for selecting the method used in SR SA-C1 for the intended application. The intended application includes but is not limited to the type of reactor and containment design and the range of POSs and accident sequence characteristics for which the method would be applied.	
SA-C3	JUSTIFY that method(s) used to adapt or modify results of a reference calculation are both applicable to the plant under review and are applied in a conservative way, or a combination of conservative and realistic ways (e.g., reference plant calculations are from a similar plant design and bounding assumptions are used).	JUSTIFY selections of modeling options in the modeling tool selected in SA-C1 (e.g., show that selected correlations or models within a computer code are appropriate) for significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.

Table 2-2.3-5 Supporting Requirements for HLR-SA-D

Severe Accident Progression Analysis calculations shall be performed as needed to support the probabilistic accident progression framework (HLR-SA-D).

Index No. SA-D	Capability Category I	Capability Category II
SA-D1	Apply the method selected in SR SA-C1 to ESTIMATE the output parameters specified in SR SA-A2 for all accident progression sequences.	Apply the method selected in SR SA-C1 to CALCULATE the output parameters specified in SR SA-A2 for all accident progression sequences.
SA-D2	When using reference plant calculations, JUSTIFY similarity in reactor and containment design between the reference plant and the plant being analyzed, as appropriate, by comparing plant-design or operating characteristics that influence the POS, condition, process, or event of interest (e.g., calculate and compare ratios of parameters that govern the calculated result, such as reactor power, coolant volume, clad metal mass, and containment heat removal capacity).	
SA-D3	ENSURE the reasonableness and acceptability of the calculated output parameters specified in SR SA-A2.	
SA-D4	JUSTIFY the end-point or termination time of severe accident calculations by providing a technical basis that conclusions drawn from the calculation would not change if the termination time was extended (e.g., confirm a safe stable state exists and fission product releases have plateaued, mission time supports late release sequences and properly interfaces with the Level 3 PRA).	
SA-D5	IDENTIFY method-specific limitations in the severe accident progression analysis.	

Table 2-2.3-6 Supporting Requirements for HLR-SA-E

Uncertainties in the Severe Accident Progression Analysis shall be characterized. Sources of model uncertainty and related assumptions shall be identified and their potential impact on the analysis understood (HLR-SA-E).

Index No. SA-E	Capability Category I	Capability Category II
SA-E1	IDENTIFY generic sources of uncertainty, the related assumptions, and reasonable alternatives of the Severe Accident Progression Analysis in a manner that supports the applicable requirements of SR PT-F4.	IDENTIFY generic and plant-specific sources of uncertainty, the related assumptions, and reasonable alternatives of the Severe Accident Progression Analysis in a manner that supports the applicable requirements of SR PT-F4.
SA-E2	IDENTIFY input parameters particular to the method applied in SR SA-D1 that impact the uncertainty of the models or assumptions defined in SR SA-E1.	PROVIDE variations in input parameters particular to the method applied in SR SA-D1 that impact the uncertainty of the models or assumptions defined in SR SA-E1.
SA-E3	CHARACTERIZE the effects of uncertainties associated with input parameters identified in SR SA-E2.	For significant accident progression sequences, CALCULATE the effects of uncertainties associated with model input parameter variations provided in SR SA-E2, using a quantitative method. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
SA-E4	For each source of model uncertainty and assumptions identified in SR SA-E1 that are not characterized or evaluated in SR SA-E3, CHARACTERIZE how the accident progression analysis results are affected.	

Table 2-2.3-7 Supporting Requirements for HLR-SA-F

The documentation of the Severe Accident Progression Analysis shall provide traceability of the work (HLR-SA-F).

Index No. SA-F	Capability Category I	Capability Category II
SA-F1	DOCUMENT the process used in the Severe Accident Progression Analysis specifying what is used as input, the applied methods, and the results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the objectives of the severe accident progression analysis and the quantitative output parameters that are used in the Severe Accident Progression Analysis *(b)* the assumptions used to perform deterministic calculations and the estimated values of input parameters *(c)* the deterministic (computational) model selected for generating the quantitative parameters *(d)* the calculations performed to support the probabilistic accident progression framework	
SA-F2	Where reference plant results are used, DOCUMENT the method used to adapt or modify the results of reference plant calculations.	DOCUMENT user-defined input data for computer codes including references to sources of information and derivations of calculated parameters. Sufficient detail is to be provided for an independent person to be able to reproduce the input data from original sources.
SA-F3	DOCUMENT the parameter uncertainty associated with inputs to the Severe Accident Progression Analysis.	
SA-F4	DOCUMENT results of calculations including the sequence of important severe accident phenomena.	
SA-F5	DOCUMENT the reasonableness of calculated results.	
SA-F6	DOCUMENT the sources of model uncertainty, related assumptions, their characterization, and reasonable alternatives associated with the Severe Accident Progression Analysis identified in SR SA-E1 and SR SA-E2.	

2-2.4 PROBABILISTIC TREATMENT OF ACCIDENT PROGRESSION AND SOURCE TERMS (PT)

2-2.4.1 Objectives

The objectives of the Probabilistic Treatment of Accident Progression and Source Terms are to establish a framework to support the systematic quantification of the potential severe accident sequences derived from Level 1 PRA core damage sequences in sufficient detail such that

(a) the accident progression framework groups radionuclide release categories

(b) branching probabilities or supporting models that support quantification of severe accident phenomena are developed

(c) branching probabilities or supporting models that support quantification of severe accident mitigation equipment reliability are developed

(d) branching probabilities or supporting models that support quantification of severe accident human actions are developed

(e) the frequencies of radionuclide release categories are calculated using appropriate models and codes

(f) uncertainties in the radionuclide release category frequencies are characterized

(g) the Probabilistic Treatment of Accident Progression and Source Terms is documented to provide traceability of the work

Table 2-2.4-1 provides the HLRs for the Probabilistic Treatment of Accident Progression and Source Terms.

Table 2-2.4-1 High Level Requirements for Probabilistic Treatment of Severe Accident Progression and Source Terms (PT)

Designator	Requirement
HLR-PT-A	An accident progression framework shall be developed that supports the grouping of severe accident sequences into radionuclide RCs (source term) .
HLR-PT-B	Branching probabilities (split fractions) or supporting models for quantitatively characterizing severe accident phenomena shall be estimated.
HLR-PT-C	Branching probabilities (split fractions) or supporting models for quantitatively characterizing the reliability of modeled equipment in the accident progression framework shall be included.
HLR-PT-D	Branching probabilities (split fractions) or supporting models for quantitatively characterizing the reliability of human actions in the accident progression framework shall be included.
HLR-PT-E	The frequencies of radionuclide RCs, using appropriate models and codes accounting for method-specific limitations and features, shall be calculated.
HLR-PT-F	Uncertainties in the frequencies of radionuclide RCs shall be defined. Sources of model uncertainty and related assumptions shall be identified and their potential impact on the estimation understood.
HLR-PT-G	The documentation of the Probabilistic Treatment of Accident Progression and Source Terms shall provide traceability of the work.

Table 2-2.4-2 Supporting Requirements for HLR-PT-A

An accident progression framework shall be developed that supports the grouping of severe accident sequences into radionuclide RCs (source term) (HLR-PT-A).

Index No. PT-A	Capability Category I	Capability Category II
PT-A1	SELECT a methodology for representing the severe accident progression that *(a)* models the combination of system responses, operator actions, and severe accident phenomena that affect the accident progression *(b)* includes a representation of the accident progression associated with each Level 1 PRA accident sequence or plant damage state (PDS) *(c)* provides a framework to support RC frequency quantification *(d)* supports the grouping of severe accident sequences into radionuclide RCs (source term)	
PT-A2	DEVELOP a logic structure using the method selected in SR PT-A1 (i.e., severe accident progression method or CET or equivalent).	
PT-A3	INCLUDE attributes of the severe accident progression in the logic structure developed in SR PT-A2. Attributes include but may not be limited to the following: *(a)* chronological treatment of events that preserves the order and approximate timeline with which severe accident progression results in radiological release to the environment *(b)* identification and probabilistic assessment of mechanisms for defeating (by failure or bypass) physical barriers to the release of radioactive material to the environment *(c)* numerical accounting of accident sequence frequency from the initiating event to the summed frequency of all end states *(d)* aggregation of individual accident progressions into groups (RCs) that have common characteristics of radiological release to the environment and a calculation of their associated frequency *(e)* consistency in the treatment of dependencies with linkages to Level 1 PRA models *(f)* credited human actions including those that could have negative impacts *(g)* treatment of source-term reduction and capture of fission products due to scrubbing in water pools, sprays, or deposition	

Table 2-2.4-2 Supporting Requirements for HLR-PT-A (Cont'd)

Index No. PT-A	Capability Category I	Capability Category II
PT-A4	INCLUDE the following attributes in the logic structure developed in SR PT-A2: *(a)* initial conditions of Level 2 analysis (output from Level 1/Level 2 PRA Interface) *(b)* discrimination of different radiological release pathways to the environment *(c)* the effects of restoration of the coolant injection function prior to RPV lower head failure *(d)* containment and RPV status at the time of core damage *(e)* accident progression phenomena that affect the evaluation of containment failure or bypass (address at least those phenomena in Table 2-2.4-9) *(f)* loss of containment integrity including time of failure and resulting leakage area and location(s) *(g)* status of containment mitigation systems including sprays, air cleanup, and ventilation systems *(h)* accident progression phenomena that may be important to specific POSs; these include phenomena such as air ingression and its effects on fuel cladding oxidation and fission product release during reactor shutdown accident sequences	
PT-A5	INCLUDE events in the logic structure developed in SR PT-A2 that represent the characteristics of severe accident progression (phenomenological events) that could generate mechanical loads and/or thermal challenges to the containment pressure boundary sufficient to cause structural failure or increased leakage or could induce a release pathway that bypasses the containment pressure boundary, including those identified in Table 2-2.4-9.	
PT-A6	JUSTIFY the exclusion of any phenomenological event characteristics identified in Table 2-2.4-9 from the model (e.g., if any of the phenomenological event characteristics that are not applicable to your plant design).	
PT-A7	INCLUDE in the logic structure developed in SR PT-A2 events that represent the positive or negative effects of mitigating actions directed by plant-specific procedures or guidelines on radionuclide release for significant accident progression sequences.	
PT-A8	JUSTIFY the exclusion in the logic structure developed in SR PT-A2 of events that reflect accident behavior within structures outside the containment pressure boundary or the response of mitigating systems outside the containment pressure boundary that affect source-term attenuation (e.g., simplified treatment of source term attenuation).	INCLUDE in the logic structure developed in SR PT-A2 events that reflect accident behavior within structures outside the containment pressure boundary or the response of mitigating systems outside the containment pressure boundary that affect source-term attenuation for significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-A9	INCLUDE in the logic structure developed in SR PT-A2 expected beneficial failures of passive SSCs.	INCLUDE in the logic structure developed in SR PT-A2 expected beneficial failures of active and passive SSCs in significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-A10	INCLUDE in the logic structure developed in SR PT-A2 the capability to determine importance measures to RCs and/or surrogate release metric for PDSs.	INCLUDE in the logic structure developed in SR PT-A2 the capability to determine importance measures to RCs and/or surrogate release metric for PDSs, individual SSCs, human actions, severe accident phenomena that contribute to significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.

Table 2-2.4-2 Supporting Requirements for HLR-PT-A (Cont'd)

Index No. PT-A	Capability Category I	Capability Category II
PT-A11	MODEL the logical dependencies between systems, components, and human actions in the Level 1 PRA and in the logic structure developed in SR PT-A2 in a manner that would result in an earlier time of containment failure and/or a larger radiological source term than expected in a realistic analysis. This includes typical support-system dependencies as well as dependencies specific to severe accidents.	MODEL the logical dependencies between systems, components, and human actions in the Level 1 PRA and in the logic structure developed in SR PT-A2 realistically for significant accident progression sequences. This includes typical support-system dependencies as well as dependencies specific to severe accidents. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-A12	ENSURE the logic structure developed in SR PT-A2 generates a conservative or a combination of conservative and realistic assessment of the frequency of accident progression sequences.	ENSURE the logic structure developed in SR PT-A2 generates a realistic assessment of accident progression frequency for significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-A13	INCLUDE events in the logic structure developed in SR PT-A2 that represent operator actions required to establish containment closure during accident sequences with an open containment pressure boundary.	
PT-A14	SPECIFY end states using the definitions and attributes of RCs described in HLR-ST-A.	
PT-A15	IDENTIFY the RC for each of the Level 2 PRA accident sequences (or equivalent).	
PT-A16	CHARACTERIZE the RCs identified in SR PT-A15.	
PT-A17	JUSTIFY categorization of non-LERF/non-LRF releases (e.g., based on release magnitude or timing); see SR ST-A3, SR ST-A4, and SR ST-A5.	

Table 2-2.4-3 Supporting Requirements for HLR-PT-B

Branching probabilities (split fractions) or supporting models for quantitatively characterizing severe accident phenomena shall be estimated (HLR-PT-B).

Index No. PT-B	Capability Category I	Capability Category II
PT-B1	USE a method or methods for developing probabilities of phenomenological events such that uncertainties can be characterized (see HLR-PT-F). If using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment.	
PT-B2	JUSTIFY the rationale for the method or methods used in SR PT-B1 (e.g., refer to results of thermal hydraulic sensitivity studies, consensus documents, or research findings including relevant experimental studies). If using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment.	
PT-B3	ESTIMATE branching probabilities (split fractions) for phenomenological events using conservative, or a combination of conservative and realistic, boundary conditions.	ESTIMATE branching probabilities (split fractions) in significant accident progression sequences for phenomenological events using realistic boundary conditions. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-B4	ESTIMATE the conditional probability of phenomenologically induced containment bypass events in a conservative, or a combination of conservative and realistic, manner.	ESTIMATE the conditional probability of phenomenologically induced containment bypass events in a realistic manner in significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-B5	COMPARE the Severe Accident Progression Analysis containment challenge output parameters analyzed in HLR-SA-D to the containment capacities analyzed in HLR-CP-C to identify those challenges that can result in containment failure.	
PT-B6	IDENTIFY accident progression sequences that have the potential for large and large/early radionuclide releases (refer to HLR-ST-A).	IDENTIFY accident progression sequences that have the potential for radionuclide release (refer to HLR-ST-A).
PT-B7	Using the comparison performed in SR PT-B5, ESTIMATE the probability of containment failure events.	
PT-B8	ASSUME that secondary containment or auxiliary building(s) do not act as an effective radionuclide barrier to radionuclide release for containment failure sequences with core damage.	ESTIMATE the probability of secondary containment or auxiliary building(s) acting as a retention location preventing or reducing radionuclide release for each accident sequence including the impact of phenomenological effects such as hydrogen combustion or external hazards; JUSTIFY the estimated probabilities (e.g., using the results of detailed analyses that include the appropriate retention processes).

Table 2-2.4-4 Supporting Requirements for HLR-PT-C

Branching probabilities (split fractions) or supporting models for quantitatively characterizing the reliability of modeled equipment in the accident progression framework shall be included (HLR-PT-C).

Index No. PT-C	Capability Category I	Capability Category II
PT-C1	INCLUDE new system models and split fractions needed to support the accident progression analysis and ENSURE new system models meet at least the Capability Category I requirements in [2-1] as applicable to the scope of the Level 2 PRA or SPECIFY a basis to support the claim of nonapplicability of any of these requirements: *(a)* Part 2–Internal Events PRA: HLR-SY-A through HLR-SY-C and HLR-DA-A through HLR-DA-E *(b)* Part 3–Internal Flood PRA: HLR-IFSN-A through HLR-IFSN-B, HLR-IFPR-A through HLR-IFPR-C, and HLR-IFQU-A though HLR-IFQU-G *(c)* Part 4–Internal Fire PRA: HLR-ES-A through HLR-ES-D, HLR-CS-A through HLR-CS-C, HLR-QLS-A through HLR-QLS-B, HLR-PRM-A through HLR-PRM-C, HLR-FSS-A through HLR-FSS-H, HLR-CF-A through HLR-CF-B, and HLR-FQ-A through HLR-FQ-G *(d)* Part 5–Seismic Event PRA: HLR-SFR-A through HLR-SFR-F and HLR-SPR-A through HLR-SPR-F *(e)* Part 7–High Wind PRA: HLR-WFR-A through HLR-WFR-I and HLR-WPR-A through HLR-WPR-F *(f)* Part 8–External Flood PRA: HLR-XFFR-A through HLR-XFFR-F and HLR-XFPR-A through HLR-XFPR-G *(g)* Part 9–Other Hazard PRA: HLR-XFR-A through HLR-XFR-B and HLR-XPR-A through HLR-XPR-E For modeled low power and shutdown POSs, ENSURE new system models meet at least the Capability Category I requirements in [2-2] as applicable to the scope of the Level 2 PRA or SPECIFY a basis to support the claim of nonapplicability of any of the requirements in [2-2].	INCLUDE new system models and split fractions needed to support the accident progression analysis and ENSURE new system models meet the Capability Category II requirements in [2-1] as applicable to the scope of the Level 2 PRA or SPECIFY a basis to support the claim of nonapplicability of any of these requirements: *(a)* Part 2–Internal Events PRA: HLR-SY-A through HLR-SY-C and HLR-DA-A through HLR-DA-E *(b)* Part 3–Internal Flood PRA: HLR-IFSN-A through HLR-IFSN-B, HLR-IFPR-A through HLR-IFPR-C, and HLR-IFQU-A though HLR-IFQU-G *(c)* Part 4–Internal Fire PRA: HLR-ES-A through HLR-ES-D, HLR-CS-A through HLR-CS-C, HLR-QLS-A through HLR-QLS-B, HLR-PRM-A through HLR-PRM-C, HLR-FSS-A through HLR-FSS-H, HLR-CF-A through HLR-CF-B, and HLR-FQ-A through HLR-FQ-G *(d)* Part 5–Seismic Event PRA: HLR-SFR-A through HLR-SFR-F and HLR-SPR-A through HLR-SPR-F *(e)* Part 7–High Wind PRA: HLR-WFR-A through HLR-WFR-I and HLR-WPR-A through HLR-WPR-F *(f)* Part 8–External Flood PRA: HLR-XFFR-A through HLR-XFFR-F and HLR-XFPR-A through HLR-XFPR-G *(g)* Part 9–Other Hazard PRA: HLR-XFR-A through HLR-XFR-B and HLR-XPR-A through HLR-XPR-E For modeled low power and shutdown POSs, ENSURE new system models meet the Capability Category II requirements in [2-2] as applicable to the scope of the Level 2 PRA or SPECIFY a basis to support the claim of nonapplicability of any of the requirements in [2-2].
PT-C2	USE analyses of system-success criteria that are applicable to the plant and meet at least the applicable Capability Category I requirements in HLR-SC-A through HLR-SC-C of Part 2 of [2-1]; for modeled low power and shutdown POSs, SATISFY at least the applicable Capability Category I requirements in HLR-LSC-A through HLR-LSC-C of Part 3 of [2-2]. SPECIFY a basis to support the claim of nonapplicability of any of these requirements.	USE analyses for system-success criteria that are applicable to the plant and meet the applicable Capability Category II requirements in HLR-SC-A through HLR-SC-C of Part 2 of [2-1]; for modeled low power and shutdown POSs, SATISFY the applicable Capability Category II requirements in HLR-LSC-A through HLR-LSC-C of Part 3 of [2-2] for significant accident progression sequences. SPECIFY a basis to support the claim of nonapplicability of any of these requirements. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.

Table 2-2.4-4 Supporting Requirements for HLR-PT-C (Cont'd)

Index No. PT-C	Capability Category I	Capability Category II
PT-C3	For Level 2 accident sequence analysis, SATISFY at least the Capability Category I requirements in HLR-AS-A through HLR-AS-C of Part 2 of [2-1]; for modeled low power and shutdown POSs, SATISFY at least the applicable Capability Category I requirements in HLR-LAS-A through HLR-LAS-C of Part 3 of [2-2]. SPECIFY a basis to support the claim of nonapplicability of any of these requirements.	For Level 2 accident sequence analysis, SATISFY the Capability Category II requirements in HLR-AS-A through HLR-AS-C of Part 2 of [2-1]; for modeled low power and shutdown POSs, SATISFY the applicable Capability Category II requirements in HLR-LAS-A through HLR-LAS-C of Part 3 of [2-2] for significant accident progression sequences. SPECIFY a basis to support the claim of nonapplicability of any of these requirements. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-C4	For system models in the Level 2 analysis, ENSURE the component mission time supports the accident progression sequence timing, as evaluated in SR SA-D4, for which the component is credited, or JUSTIFY a shorter component mission time (e.g., why the component fulfills its mission even if it only operates for a shorter period of time).	
PT-C5	USE conservative or a combination of conservative and realistic treatments of adverse environmental impacts for assessing equipment survivability for equipment inside containment.	USE realistic treatments of adverse environmental impacts for assessing equipment survivability for equipment inside containment that would be risk significant if assumed not to survive given severe accident conditions. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-C6	USE conservative or a combination of conservative and realistic treatments of adverse environmental impacts for assessing equipment survivability for equipment outside containment (including equipment credited from other units on site).	USE realistic treatments of adverse environmental impacts for assessing equipment survivability for equipment outside containment (including equipment credited from other units on site) that would be risk significant if assumed not to survive given severe accident conditions (including containment failure). For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-C7	USE a conservative evaluation of secondary side isolation capability for accident sequences caused by SGTR (if applicable). If generic analyses generated for similar plants are used, JUSTIFY applicability to the plant being evaluated (e.g., similar isolation capability and similar containment designs).	PERFORM a realistic secondary side isolation capability analysis for significant accident progression sequences caused by SGTR (if applicable). For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.

Table 2-2.4-4 Supporting Requirements for HLR-PT-C (Cont'd)

Index No. PT-C	Capability Category I	Capability Category II
PT-C8	PERFORM a conservative analysis of severe accident-induced SGTR probability (if applicable) that includes plant-specific procedures and design features.	PERFORM a realistic analysis of severe accident-induced SGTR (if applicable) that includes plant-specific procedures and design features that could impact tube failure probability assessment in significant accident progression sequences. SELECT failure probabilities based on: *(a)* RCS and steam generator post-accident conditions sufficient to describe the important risk outcomes *(b)* secondary side conditions including plant-specific treatment of steam generator safety valves and atmospheric dump valves For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-C9	PERFORM containment isolation system analysis in a conservative or a combination of conservative and realistic manner. In the containment isolation system analyses, INCLUDE both the analysis of the failure of containment isolation systems to perform properly and the status of safety systems that do not have automatic isolation provisions including consideration of status of systems for the modeled POSs, failures of penetrations, seals, and hatches plus pre-existing failures.	PERFORM containment isolation system analysis in a realistic manner for the significant accident progression sequences. In the containment isolation system analyses, INCLUDE both the analysis of the failure of containment isolation systems to perform properly and the status of safety systems that do not have automatic isolation provisions including consideration of status of systems for the modeled POSs, failures of penetrations, seals, and hatches plus pre-existing failures. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-C10	PERFORM a conservative or a combination of conservative and realistic interfacing system failure analysis for accident progression sequences resulting in a radionuclide release due to failure of the interfacing systems.	PERFORM a realistic interfacing system failure analysis for the significant accident progression sequences resulting in a radionuclide release due to failure of the interfacing systems. In the interfacing system failure analysis, include behavior of piping relief valves, pump seals, heat exchangers at applicable temperature and pressure conditions, and consideration of modeled POSs. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-C11	JUSTIFY any credit taken for continued successful operation of mitigation systems under adverse environments assessed in SR PT-C5 and SR PT-C6. Justification may include washdown of aerosols that may allow continued operation of plant equipment, consideration of specific component failure modes, or expected component survivability (e.g., response to temperature and pressure spikes) during time frame of component operation.	
PT-C12	JUSTIFY any credit taken for beneficial failures in SR PT-A9 (e.g., credit for beneficial hot-leg failure could be based on results of realistic thermal hydraulic analyses).	

Table 2-2.4-5 Supporting Requirements for HLR-PT-D

Branching probabilities (split fraction) or supporting models for quantitatively characterizing the reliability of human actions in the accident progression framework shall be included (HLR-PT-D).

Index No. PT-D	Capability Category I	Capability Category II
PT-D1	INCLUDE new HFEs as needed to support the accident progression analysis and ENSURE new HFEs meet at least the Capability Category I requirements in [2-1] as applicable to the scope of the Level 2 PRA or SPECIFY a basis to support the claim of nonapplicability of any of these requirements: *(a)* Part 2–Internal Events PRA: HLR-HR-E through HLR-HR-I *(b)* Part 3–Internal Flood PRA: HLR-IFHR-A through HLR-IFHR-E *(c)* Part 4–Internal Fire PRA: HLR-FHR-A through HLR-FHR-E *(d)* Part 5–Seismic Event PRA: HLR-SPR-D *(e)* Part 7–High Wind PRA: HLR-WPR-D *(f)* Part 8–External Flood PRA: HLR-XFPR-E *(g)* Part 9–Other Hazard PRA: HLR-XPR-C For modeled low power and shutdown POSs, ENSURE new HFEs meet at least the Capability Category I requirements in [2-2] as applicable to the scope of the Level 2 PRA or SPECIFY a basis to support the claim of nonapplicability of any of the requirements in [2-2].	INCLUDE new HFEs as needed to support the accident progression analysis and ENSURE new HFEs meet the Capability Category II requirements in [2-1] as applicable to the scope of the Level 2 PRA or SPECIFY a basis to support the claim of nonapplicability of any of these requirements: *(a)* Part 2–Internal Events PRA: HLR-HR-E through HLR-HR-I *(b)* Part 3–Internal Flood PRA: HLR-IFHR-A through HLR-IFHR-E *(c)* Part 4–Internal Fire PRA: HLR-FHR-A through HLR-FHR-E *(d)* Part 5–Seismic Event PRA: HLR-SPR-D *(e)* Part 7–High Wind PRA: HLR-WPR-D *(f)* Part 8–External Flood PRA: HLR-XFPR-E *(g)* Part 9–Other Hazard PRA: HLR-XPR-C For modeled low power and shutdown POSs, ENSURE new HFEs meet the Capability Category II requirements in [2-2] as applicable to the scope of the Level 2 PRA or SPECIFY a basis to support the claim of nonapplicability of any of the requirements in [2-2].
PT-D2	ESTIMATE, in a conservative manner, the probability of applicable HFEs following the onset of core damage included in SR PT-A7. It is acceptable to assume severe accident management guideline (SAMG) actions are not successfully implemented.	ESTIMATE the realistic probability of applicable HFEs following the onset of core damage included in SR PT-A7 using applicable HRA methods that include the impact of structural changes in organizational behavior as well as the time required and the time available to perform the action for significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-D3	If crediting repair, ENSURE the credit given is conservative.	If crediting repair, ENSURE that plant conditions do not preclude repair and actuarial data exists from which to estimate the repair failure probability as required by SR SY-A24 and SR DA-C15 in Part 2 of [2-1]. Credit for off-site power recovery based on generic data applicable to the plant is acceptable.
PT-D4	For multiple human actions in the same accident progression sequence PERFORM an analysis of HFE dependency and at least meet the applicable Capability Category I requirements of Part 2 of [2-1] or SPECIFY a basis to support the claim of nonapplicability of any of these requirements; specifically, SR HR-G7, SR HR-G8, SR HR-G9, SR HR-H4 and SR QU-C2 of [2-1].	For multiple human actions in the same accident progression sequence PERFORM an analysis of HFE dependency and meet the applicable Capability Category II requirements of Part 2 of [2-1] or SPECIFY a basis to support the claim of nonapplicability of any of these requirements; specifically, SR HR-G7, SR HR-G8, SR HR-G9, SR HR-H4 and SR QU-C2 of [2-1].

Table 2-2.4-5 Supporting Requirements for HLR-PT-D (Cont'd)

Index No. PT-D	Capability Category I	Capability Category II
PT-D5	INCLUDE the impact of environmental conditions when estimating the probability of applicable HFEs in a conservative or a combination of conservative and realistic manner.	INCLUDE the impact of environmental conditions when estimating the probability of applicable HFEs in a realistic manner based on probabilistic and/or deterministic analyses for the significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-D6	JUSTIFY treatment of fission product scrubbing as a basis for reducing local levels of radioactivity to levels that support human actions (e.g., cite relevant experimental evidence or results of deterministic calculations for the decontamination factor used).	
PT-D7	JUSTIFY treatment of the complement of HEPs following the onset of core damage (i.e., the probability of successful operator actions following the onset of core damage) (e.g., demonstrate that excessive conservatism, such as the use of screening values, was not applied to an HEP in cases where this would lead to a less severe end state).	

Table 2-2.4-6 Supporting Requirements for HLR-PT-E

The frequencies of radionuclide RCs, using appropriate models and codes accounting for method-specific limitations and features, shall be calculated (HLR-PT-E).

Index No. PT-E	Capability Category I	Capability Category II
PT-E1	CALCULATE the frequency of each RC (as defined in HLR-ST-A) consistent with the methods prescribed in HRL-PT-A, HLR-PT-B, HLR PT-C, and HLR-PT-D.	
PT-E2	EVALUATE dependencies introduced by common physical parameters involved in multiple CET top events (or equivalent) in a conservative or a combination of conservative and realistic manner.	EVALUATE dependencies introduced by common physical parameters involved in multiple CET top events (or equivalent) in a manner that provides for a realistic estimate of the frequencies of significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-E3	USE a quantification methodology that accurately models the complementary "success" state where high failure probability events are included in the Level 2 PRA logic model.	
PT-E4	SPECIFY the criteria used for the term "high failure probability" in SR PT-E3 and provide a basis for the criteria.	
PT-E5	COMPARE the end state frequencies in the Level 2 analysis to the corresponding input frequency from the Level 1/Level 2 PRA Interface and EXPLAIN any discrepancies between the sum of the CDF contributors and the total Level 2 end state frequencies.	
PT-E6	IDENTIFY any method-specific limitations in the quantitative results including those that arise from the effects of high failure probability events.	
PT-E7	PERFORM a frequency truncation study to demonstrate the degree of convergence for significant RCs consistent with SR QU-B3 in Part 2 of [2-1].	
PT-E8	SELECT appropriate truncation limits for accident sequences (or cutsets) to ensure the proper incorporation of frequencies and dependencies in each RC consistent with SR QU-B2 in Part 2 of [2-1].	

Table 2-2.4-6 Supporting Requirements for HLR-PT-E (Cont'd)

Index No. PT-E	Capability Category I	Capability Category II
PT-E9	EVALUATE importance measures to RCs and/or surrogate release metric for PDSs.	EVALUATE importance measures to RCs and/or surrogate release metric for PDSs, individual SSCs, human actions, severe accident phenomena that contribute to significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
PT-E10	REVIEW the importance measures evaluated in SR PT-E9 to ensure that they are consistent with expected results and understand or reconcile the reason for any unexpected results.	
PT-E11	REVIEW a sample of the significant accident progression sequences/cutsets for each RC sufficient to determine that the logic of the cutset or sequence is correct.	
PT-E12	REVIEW the results of the PRA for each RC for modeling consistency (e.g., event sequence model consistency with systems models and success criteria) and operational consistency (e.g., plant configuration, procedures, and plant-specific and industry experience).	
PT-E13	REVIEW results for each RC to ensure that the flag event settings, mutually exclusive event rules, and recovery rules (if applicable) yield logical results.	
PT-E14	COMPARE results to those from similar plants if information from similar plants is available.	COMPARE results to those from similar plants if information from similar plants is available and JUSTIFY causes for differences (e.g., explain why one RC is a large contributor for one plant and not another).
PT-E15	REVIEW a sampling of accident progression sequences that are not significant to determine they are reasonable and have physical meaning.	

Table 2-2.4-7 Supporting Requirements for HLR-PT-F

Uncertainties in the frequencies of radionuclide release categories shall be defined. Sources of model uncertainty and related assumptions shall be identified and their potential impact on the estimation understood (HLR-PT-F).

Index No. PT-F	Capability Category I	Capability Category II
PT-F1	CHARACTERIZE the uncertainty range for branching probabilities (split fractions) developed under HLR-PT-A, HLR-PT-B, HLR-PT-C, and HLR-PT-D. Expert judgment may be used. If using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment.	CALCULATE the uncertainty range for the parameter estimates of significant basic events used to represent branching probabilities (split fractions) developed under HLR-PT-A, HLR-PT-B, HLR-PT-C, and HLR-PT-D. Expert judgment may be used to estimate the uncertainty range for branching probabilities (split fractions) where alternate approaches are impractical. If using expert judgment, SATISFY the requirements of Section 1-4.2, Use of Expert Judgment. For basic events that are not significant, ENSURE the requirement for CC-I is met.
PT-F2	ESTIMATE the parameter uncertainty on the frequency of significant RC(s) providing a basis for the estimate consistent with the characterization of parameter uncertainties.	CALCULATE the parameter uncertainty on the frequency of significant RC(s); INCLUDE treatment of state-of-knowledge correlation if applicable. For RC(s) that are not significant, ENSURE the requirement for CC-I is met.
PT-F3	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives of the Probabilistic Treatment of Accident Progression and Source Terms in a manner consistent with SR PT-F4.	
PT-F4	ASSESS the effects of individual sources of model uncertainty, related assumptions and combinations of interest identified for each technical element (including those transferred in SR L1-B8).	

Table 2-2.4-8 Supporting Requirements for HLR-PT-G

The documentation of the Probabilistic Treatment of Accident Progression and Source Terms shall provide traceability of the work (HLR-PT-G).

Index No. PT-G	Capability Category I	Capability Category II
PT-G1	DOCUMENT the process used in the Probabilistic Treatment of Accident Progression and Source Terms specifying what is used as input, the applied methods, and results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* accident progression framework and radionuclide release (source term) categories *(b)* branching probabilities (split fractions) or supporting models for quantitatively characterizing severe accident phenomena and their bases *(c)* branching probabilities (split fractions) or supporting models for quantitatively characterizing the reliability of modeled equipment in the accident progression framework and their bases *(d)* branching probabilities (split fractions) or supporting models for quantitatively characterizing the reliability of human actions in the accident progression framework and their bases *(e)* the frequencies of radionuclide release categories, including method-specific limitations and features of models and codes used *(f)* the Level 2 PRA quantification process and results including: *(1)* general description of the quantification process *(2)* the comparison of severe accident challenges and the containment performance *(3)* the treatment of high failure probability events in the Level 2 PRA logic model quantification *(4)* the process and results for establishing the truncation values for final quantification *(5)* the parameter uncertainty on the frequency of significant RC(s) *(6)* records of the cutset review process *(7)* importance measure results *(8)* comparison of results to similar plants including causes for differences	
PT-G2	DOCUMENT the sources of model uncertainty, related assumptions, their characterization and reasonable alternatives associated with the Probabilistic Treatment of Accident Progression and Source Terms identified in HLR-PT-F.	

Table 2-2.4-9 Containment Failure or Bypass Events to Be Assessed in the Development of a Level 2 PRA

Contributor	Containment Design [Note (1)]				
	AP-1000, U.S. EPR, U.S. APWR, Large Dry & Subatmospheric	Ice Condenser	BWR Mark I	ABWR, ESBWR BWR Mark II	BWR Mark III
Containment isolation failure	x	x	x	x	x [Note (2)]
Containment Bypass					
(a) ISLOCA	x	x	x	x	x
(b) SGTR	x	x	. . .	. . .	. . .
(c) Induced SGTR	x	x	. . .	. . .	. . .
(d) Isolation condenser tube rupture			x (if applicable)	x (if applicable)	. . .
Energetic containment failures					
(a) High pressure melt ejection (HPME)	x	x	x	x	x
(b) Hydrogen combustion	x	x	x [Note (3)]	x [Note (3)]	x
RPV vertical displacement due to blowdown forces [Note (4)]	x	x	. . .	. . .	. . .
Core debris impingement [Note (5)]	x	x	x	. . .	. . .
Steam explosion [Note (6)]	x	x	x	x	x
Shell melt-through	. . .	. . .	x (if applicable)	x (if applicable)	. . .
Pressure suppression bypass [Note (7)]	. . .	x	x	x	x
RPV and/or containment venting	x (if applicable)	x (if applicable)	x	x	x
Vacuum breaker failure	. . .	. . .	x	x	x
Hydrodynamic loads under severe accident conditions	. . .	. . .	x	x	x
Overpressure failure due to increases in quasi-static pressure (i.e., steam and non-condensable gas content) combined with increased atmosphere temperature	x	x	x	x	x
Mechanical and electrical penetration failure	x	x	x	x	x
Containment leakage at hatches, past degraded penetration seals as well as pre-existing liner leakage	x	x	x	x	x
BMMT	x	x	x	x	x

GENERAL NOTE: Combinations of contributors may also be considered where appropriate. For example, in a BWR Mark I or II, the combination of containment flooding and containment venting may be considered.

NOTES:
(1) The containment failure and bypass events listed in this table are not intended to be prescriptive or complete. They are minimal starting points for the analysis of reactor/containment configurations with some Level 2 PRA experience (industry and/or NRC).
(2) Drywell (DW) isolation failure.
(3) Combustion within the containment might be precluded during at-power operation when the containment is inerted. The consideration of combustible gases in reactor buildings and auxiliary buildings outside of containment may also introduce adverse effects on the ability to mitigate severe accidents. These effects are noted here, but this table is constructed only to address containment failure or bypass events.
(4) This failure mode is caused by the upward reaction forces accompanying RPV lower head failure at high pressure. Displacement of the RPV and attached piping can cause damage to piping penetrations and other containment structures.
(5) Refers to direct contact between molten core debris and a thin-walled (steel) containment shell.
(6) The probability of a steam explosion challenge is generally low.
(7) Ice bed bypass for ice condensers and suppression pool bypass for BWRs.

2-2.5 SOURCE TERM ANALYSIS (ST)

2-2.5.1 Objectives

The objectives of the Source Term Analysis are to calculate a unique source term to the environment for each radionuclide RC in such a way that:

(a) Release categories are defined based on the Source Term Analysis.
(b) Source terms are determined.
(c) Uncertainties in radionuclide release/transport phenomena are characterized.
(d) The Source Term Analysis is documented to provide traceability of the work.

Table 2-2.5-1 provides the HLRs for the Source Term Analysis.

Table 2-2.5-1 High Level Requirements for Source Term Analysis (ST)

Designator	Requirement
HLR-ST-A	The Source Term Analysis shall define RCs.
HLR-ST-B	The source terms shall be determined.
HLR-ST-C	Uncertainties in the radionuclide release/transport phenomena analysis shall be characterized. Sources of model uncertainty and related assumptions shall be identified and their potential impact on the analysis understood.
HLR-ST-D	The documentation of the Source Term Analysis shall provide traceability of the work.

Table 2-2.5-2 Supporting Requirements for HLR-ST-A

The Source Term Analysis shall define release categories (HLR-ST-A).

Index No. ST-A	Capability Category I	Capability Category II
ST-A1	SPECIFY attributes of radionuclide RCs (source term bins) in terms of sequence characteristics that affect the source term.	
ST-A2	DEFINE the RCs using the attributes specified by SR ST-A1. As a minimum, select the release magnitude to characterize RCs.	
ST-A3	DEFINE surrogate release metrics used in the analysis.	
ST-A4	SPECIFY the attributes used in the development of surrogate release metrics. INCLUDE in these attributes a description of the manner in which events that affect the effectiveness of off-site public response (such as external hazards) is treated.	
ST-A5	By using the attributes specified in SR ST-A1, SPECIFY the particular combination of attributes that uniquely identifies the surrogate release metrics defined in SR ST-A3. For accident progression sequences that do not have surrogate release metrics, SPECIFY a simplified set of source term characteristics.	By using the attributes specified in SR ST-A1, SPECIFY the combination of attributes that uniquely identifies the common characteristics of accident sequences leading to each RC.
ST-A6	DEMONSTRATE that combinations of attributes defined in SR ST-A5 lead to a complete set of RCs that supports the scope of the Level 2 PRA.	
ST-A7	JUSTIFY the time period for which radiological releases to the environment are considered in the characterization of RC (e.g., confirm the cumulative release has stabilized); INCLUDE consideration of the analysis termination time defined in SR SA-D4.	

Table 2-2.5-3 Supporting Requirements for HLR-ST-B

The source terms shall be determined (HLR-ST-B).

Index No. ST-B	Capability Category I	Capability Category II
ST-B1	ESTIMATE source terms using available and applicable existing generic or reference plant source term analyses or a combination of plant-specific and generic or reference plant source term analyses. JUSTIFY applicability of generic analyses; for example, by considering similarities and differences in: *(a)* reactor and containment design features in the plant or facility being studied versus the reference plant or facility *(b)* accident sequence characteristics for which the generic or reference source term was developed versus the sequences for which the source term will be applied in the PRA *(c)* operating fission product transport and attenuation mechanisms represented in the generic or reference source term(s) versus those expected to operate in the sequences in the PRA characterized by the selected source term	CALCULATE plant-specific source terms according to the requirements delineated in HLR-SA-A, HLR-SA-B, HLR-SA-C, and HLR-SA-D to quantify the source term characteristics defined in SR ST-A2 for the attributes specified in SR ST-A5 to represent each RC using models, empirical correlations, and values of parameters based on a best-estimate representation of the physical processes of fission product transport from its source, through release pathways that involve the RCS and containment, to the environment. When conservative assumptions, values for input parameters, or models are used, JUSTIFY their use (e.g., no state-of-practice or consensus model exists, impact on release profile is small or choices represent potential limitations of analytical tools) and IDENTIFY the impacts on the results of intentionally conservative or bounding inputs or models.
ST-B2	IDENTIFY the attributes used to select a representative accident sequence within each RC that include accident sequence frequency as well as the magnitude and timing of fission product release to the environment.	
ST-B3	SELECT a sequence within each RC that represents a conservative source term for sequences within the RC (i.e., the representative sequence generates a larger and/or earlier release of similar magnitude than would result from the other sequences within the RC).	SELECT a sequence within each RC that provides a realistic representation of the source term for significant accident progression sequences. For accident progression sequences that are not significant, ENSURE the requirement for CC-I is met.
ST-B4	CHARACTERIZE the conservatisms in the sequence selected in SR ST-B3.	CHARACTERIZE the range of source terms for accident sequences within each RC selected in SR ST-B3.
ST-B5	JUSTIFY any credit taken for fission product scrubbing as a basis for reducing releases through a containment bypass pathway (e.g., cite relevant experimental evidence or results of deterministic calculations for the decontamination factor used).	
ST-B6	IDENTIFY method-specific limitations in the Source Term Analysis.	

Table 2-2.5-4 Supporting Requirements for HLR-ST-C

Uncertainties in the radionuclide release/transport phenomena analysis shall be characterized. Sources of model uncertainty and related assumptions shall be identified and their potential impact on the analysis understood (HLR-ST-C).

Index No. ST-C	Capability Category I	Capability Category II
ST-C1	IDENTIFY the sources of model uncertainty, the related assumptions, and reasonable alternatives of the Source Term Analysis in a manner that supports the applicable requirements of SR PT-F4.	
ST-C2	IDENTIFY uncertain parameters influencing source terms for the representative sequences.	
ST-C3	For representative sequences, CHARACTERIZE the effects of uncertainties associated with input parameters identified in SR ST-C2.	For representative sequences CALCULATE the effects of uncertainties associated with model input parameters identified in SR ST-C2 using a quantitative method for significant RCs. For RCs that are not significant, ENSURE the requirement for CC-I is met.
ST-C4	For each source of model uncertainty and assumption identified in SR ST-C1 that are not investigated in SR ST-C3, CHARACTERIZE how the Source Term Analysis is affected.	

Table 2-2.5-5 Supporting Requirements for HLR-ST-D

The documentation of the Source Term Analysis shall provide traceability of the work (HLR-ST-D).

Index No. ST-D	Capability Category I	Capability Category II
ST-D1	DOCUMENT the process used in the Source Term Analysis specifying what is used as input, the applied methods, and results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* RC definitions *(b)* the results of source term calculations	
ST-D2	DOCUMENT attributes of radionuclide RCs specified in SR ST-A1 and combinations specified in SR ST-A5 for each RC included in the scope of the Level 2 PRA.	
ST-D3	DOCUMENT the definition and associated justification of any surrogate release metrics used in addition to the RCs (such as LR and/or large early release).	
ST-D4	DOCUMENT the manner in which a surrogate release metric (defined in SR ST-A3) is related to RCs defined in SR ST-A2.	
ST-D5	DOCUMENT the sources of model uncertainty, related assumptions, their characterization, and reasonable alternatives associated with the Source Term Analysis identified in SR ST-C1, SR ST-C2, SR ST-C3, and SR ST-C4.	

2-2.6 EVALUATION AND PRESENTATION OF RESULTS (ER)

2-2.6.1 Objectives

The objectives of the Evaluation and Presentation of Results are to present the evaluation and results in such a way that

(a) The Evaluation and Presentation of Results identifies significant contributors, characterizes uncertainties, and identifies limitations adequately.

(b) The Evaluation and Presentation of Results is documented to provide traceability of the work.

Table 2-2.6-1 provides the HLRs for the Evaluation and Presentation of Results.

Table 2-2.6-1 High Level Requirements for Evaluation and Presentation of Results (ER)

Designator	Requirement
HLR-ER-A	The Evaluation and Presentation of Results shall identify the significant contributors to Level 2 PRA, the quantitative and qualitative process used to characterize uncertainty, and the limitations in the analysis that could impact the applicability of the results.
HLE-ER-B	The documentation of the Evaluation and Presentation of Results shall provide traceability of the work.

Table 2-2.6-2 Supporting Requirements for HLR-ER-A

The Evaluation and Presentation of Results shall identify the significant contributors to Level 2 PRA, the quantitative and qualitative process used to characterize uncertainty, and the limitations in the analysis that could impact the applicability of the results (HLR-ER-A).

Index No. ER-A	Capability Category I	Capability Category II
ER-A1	IDENTIFY significant contributors (e.g., initiating events, Level 1 PRA accident sequences, basic events, cutsets, PDSs, accident progression sequences, phenomena, containment challenges, containment failure modes, modeled plant operating states, and source term categories) and sources of model uncertainty and related assumptions to each significant RC.	
ER-A2	IDENTIFY the process used to characterize the potential combined impacts of sources of uncertainty that have been characterized quantitatively and those that have been characterized qualitatively in the Level 2 PRA analysis (see SR L1-B10, HLR-CP-D, HLR-SA-E, HLR-PT-F, and HLR-ST-C).	
ER-A3	IDENTIFY limitations in the scope and level of detail of the Level 2 PRA analysis that could impact potential applications (see SR CP-C5, SR SA-D5, SR PT-E5, SR PT-E6, and SR ST-B6).	
ER-A4	IDENTIFY limitations arising from the modeling assumptions made in the analysis and/or processes, phenomena, or actions excluded from the analysis.	

Table 2-2.6-3 Supporting Requirements for HLR-ER-B

The documentation of the Evaluation and Presentation of Results shall provide traceability of the work (HLR-ER-B).

Index No. ER-B	Capability Category I	Capability Category II
ER-B1	DOCUMENT the process used in the Evaluation and Presentation of Results, specifying what is used as input, the applied methods, and results. Address the following, as well as other details needed to fully document how the set of SRs are satisfied: *(a)* the fraction of the Level 1 PRA CDF captured in the Level 2 PRA analysis and the cause of any reduction in the scope of the Level 2 PRA analysis from 100% CDF *(b)* limitations arising from the scope, level of detail, or modeling assumptions	
ER-B2	DOCUMENT significant contributors (e.g., initiating events, Level 1 PRA accident sequences, basic events, cutsets, PDSs, accident progression sequences, phenomena, containment challenges, containment failure modes, modeled POSs, and source term categories) and sources of model uncertainty and related assumptions to each significant RC.	

Section 2-3
References

[2-1] ASME/ANS RA-S-1.1-2024. Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications. The American Society of Mechanical Engineers and the American Nuclear Society.

[2-2] ANS/ASME-58.22-2014. Requirements for Low Power and Shutdown Probabilistic Risk Assessment. The American Nuclear Society and the American Society of Mechanical Engineers (Trial Use).

NONMANDATORY APPENDIX 2-A
EXPLANATORY NOTES REGARDING APPLICATION AND REVIEW OF THE LEVEL 2 SUPPORTING REQUIREMENTS

2-A.1 BACKGROUND AND OVERVIEW

This NMA provides notes and general explanatory material tied to specific SRs as stated in Part 2 of this Standard. The material contained in this Appendix is nonmandatory and, as such, does not establish new requirements; rather, the material is intended to clarify the intent of an SR, explain jargon that might be used in an SR, and/or provide examples of analysis approaches that would meet the intent of the SR.

Note that a useful collection of defined severe accident-related terms can be found in NUREG-2122 [2-A-1].

2-A.2 COMMENTARY TO LEVEL 2 TECHNICAL ELEMENTS AND REQUIREMENTS

2-A.2.1 Commentary to Level 1/Level 2 PRA Interface (L1)

One such structure that would meet the objectives of the Level 1/Level 2 PRA Interface and also provides a convenient transition point for summarizing the contributors to CDF is to consolidate or group accident sequences (or individual cutsets) from the Level 1 PRA in a manner that reduces the number of unique scenarios for evaluation while preserving the initial and boundary conditions to the analysis of plant response (i.e., PDS or equivalent).

This Section provides commentary for SRs contained in Table 2-2.1-2, Table 2-2.1-3, and Table 2-2.1-4 of Part 2 of this Standard. The following tables provide the commentary or additional material for the SRs helpful in understanding the intent of the requirement.

Table 2-A.2.1-1 Commentary to High Level Requirements for the Level 1/Level 2 PRA Interface (L1)

Designator	Commentary
HLR-L1-A	No commentary provided.
HLR-L1-B	No commentary provided.
HLR-L1-C	No commentary provided.

Table 2-A.2.1-2 Commentary to Supporting Requirements for HLR-L1-A

Index No. L1-A	Commentary
L1-A1	The purpose of this SR is simply to identify physical characteristics that are important to the Level 2 analysis. These characteristics may be included directly in the Level 1 PRA or its supporting calculations, in bridge trees, or will be modeled to support the Level 2 analysis specifically. *In defining* containment integrity (e.g., open, intact, vented, bypassed, or failed) the following references may be helpful: *(a)* "Nuclear Power Plant Response to Severe Accidents," IDCOR Technical Summary Report, Technology for Energy Corp. 1984 [2-A-2] *(b)* NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990 [2-A-3] *(c)* NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," December 1997 [2-A-4] *(d)* EPRI Report 1022186, "Technical Foundation of Reactor Safety: Knowledge Base for Resolving Severe Accident Issues," Electric Power Research Institute, Rev. 1 (2010) [2-A-5] *(e)* Seong, C., et al., "Analysis of the Technical Status of Multi-Unit Risk Assessment in Nuclear Power Plants," Nuclear Engineering and Technology, Volume 50, Issue 3, April 2018 [2-A-6] *(f)* NUREG/CR-6595 (Rev. 1), "Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events," U.S. Nuclear Regulatory Commission (2004) [2-A-7] In item SR L1-A1*(c)*, note that hydrogen mitigation systems may include igniters or hydrogen recombiners (passive or active). In evaluating item SR Ll-A1*(e)*, note that loss of containment integrity or containment bypass prior to the onset of core damage could result either as a direct consequence of the initiating event (e.g., ISLOCA, seismic event, aircraft crash, or SGTR), as a consequence of plant response to certain accident sequences (e.g., intentional containment venting to compensate for a loss of containment heat removal), or be a characteristic of the plant configuration during shutdown. The status of containment isolation should also be considered as part of containment integrity. When considering item SR L1-A1*(i)*, recognize that for LPSD accident sequences, fuel conditions within the reactor vessel during or after refueling operations (e.g., shutdown accident sequences) span a wider range of physical states (e.g., number of exposed fuel assemblies and average burnup), which influence the initial decay heat levels and in-core fission product inventories. Also, recognize that for at-power accidents, the initial state of fuel can also vary; for example, fuel conditions at end-of-cycle may be assumed to maximize fission product inventories. When identifying plant-specific issues that may influence the interface between Level 1 PRA and Level 2 PRA severe accident progression analysis note that the plant specific issues that influence the Level 1/Level 2 PRA Interface may consist of items that are formally addressed in supporting analyses or may be based on the judgment of the analysist. Examples of plant-specific issues that may exist and influence the interface of the Level 1 PRA and Level 2 PRA severe accident progression analysis include the following: *(a)* Steam tunnel configuration in a BWR may influence reactor building environmental conditions given an un-isolated break outside containment in the steam tunnel. *(b)* Hard pipe containment vent paths may have configurations that use active systems to isolate connections to other systems or buildings to avoid discharge of combustible mixtures to unwanted locations. *(c)* Plant-specific variations in safety relief valve (SRV) design may influence the pressure of the RPV when SRVs are operating to depressurize the RPV (e.g., valves with pneumatic or electro-magnetic operators). *(d)* Plant-specific alignments that allow for the containment spray function to be supported by the use of portable pumps (fire trucks may also be used for delayed spray operations if properly sized). Note spray headers may not be effective if spray pressure and flow are insufficient. Identification of plant-specific characteristics that influence the severe accident progression may occur during the development of all aspects of the Level 2 PRA. Supporting analyses for this identification may include deterministic calculations using computer codes or hand calculations.

2-A.2.1-2 Commentary to Supporting Requirements for HLR-L1-A (Cont'd)

Index No. L1-A	Commentary
L1-A2	Examples of how accident sequence characteristics can impact physical characteristics identified in SR L1-A1 include the following: *(a)* Transients can result in high RCS pressure. *(b)* LOCAs usually can result in lower RCS pressure. *(c)* ISLOCAs and SGTRs can result in containment bypass. *(d)* Stuck-open steam generator secondary safety valve(s) can open a path to the environment following failure of steam generator (SG) tubes. *(e)* External hazards can cause damage to accident mitigation resources.
L1-A3	No commentary provided.
L1-A4	No commentary provided.
L1-A5	Possible approaches could include a "large event tree," which represents a single, contiguous event sequence model similar to those documented in NUREG-1150 [2-A-3] or a set of linked event trees (e.g., a main event tree supported by small, supporting, or decomposition event trees). An example of the "small event tree, linked fault tree" approach is given in NSAC-159 (Vols. 1–3) [2-A-8].

Table 2-A.2.1-3 Commentary to Supporting Requirements for HLR-L1-B

Index No. L1-B	Commentary
L1-B1	The Level 1 PRA is used as the starting point for the development of the Level 2 PRA. Adequate scope of the PRA refers to the initiating events, hazards, and operational modes addressed in the Level 1 PRA that are that are intended to be carried forward into Level 2 PRA model. For example, if it is intended to build an at-power Level 2 fire PRA, a documented and peer-reviewed at-power Level 1 fire PRA would be needed. All significant deficiencies found in the peer review and any other exceptions for the Level 1 PRA should have been properly resolved. The definition of significant deficiency needs to be considered in the context of the regulatory framework (i.e., outside of this Standard and on a country-by-country basis). In the United States, the PRA peer-review guidance indicates that a Finding-level observation impacts the technical adequacy of the PRA and is therefore a significant deficiency. Note that significant is in this context is not to be strictly intended as risk significant. For the purposes of constructing a Level 2 PRA, documentation findings should only be considered significant deficiencies if they impact the ability to properly build and document the Level 2 PRA.
L1-B2	Examples of dependencies include the following: *(a)* initiator and support system dependencies *(b)* prior equipment failures *(c)* operator action dependencies (including available time and resource constraints) *(d)* functional dependencies (including degraded plant conditions) and common cause dependencies
L1-B3	Example methods to transfer dependencies include the following: *(a)* treatment in Level 2 PRA *(b)* expanding Level 1 PRA *(c)* construction of a bridge tree *(d)* transfer of the information via PDSs *(e)* a combination of the above methods With regard to item *(b)* above, expanding the Level 1 PRA refers to increasing the analysis scope to incorporate the evaluation and disposition of containment or severe accident mitigation systems within the Level 1 PRA for use in the Level 2 PRA. With regard to item *(d)* above, PDS mapping may include fully linking the Level 1 and Level 2 PRA models. The specific characteristics of a PDS and the reasons they are important to the Level 2 PRA vary among reactor and containment designs. For example, high RCS pressure (at the time of reactor vessel lower head failure) can be important due to the potential for high-pressure melt ejection; however, it can also be important for creating the necessary conditions for an induced SGTR and can affect fission product deposition within the RCS. Additional examples of PDS characteristics are provided in "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants," SSG-4, International Atomic Energy Agency (IAEA), 2010 [2-A-9].

Table 2-A.2.1-3 Commentary to Supporting Requirements for HLR-L1-B (Cont'd)

Index No. L1-B	Commentary	
L1-B4	The status of certain systems that may not be relevant for a specific Level 1 accident sequence may be important in the Level 2 PRA [e.g., the low pressure injection (LPI) system model may not be necessary for high pressure core damage sequences; however, the success or failure of the LPI system may need to be known post-core damage]. This can be included in the Level 1/Level 2 PRA Interface as an element in a bridge tree or modeled directed in the CET or equivalent. This SR may be accomplished by using numerical quantification of success probability, complementary logic, or a delete term approximation (or a combination of these methods) and includes the treatment of transfers among event trees where the "successes" may not be transferred between event trees. Success logic for accident sequences refers to the logic included in a Boolean model that represents the "success branch" of event trees and reflects those events that are logically excluded from the end state cutsets for sequences modeled on the branches of the Level 1 PRA event tree. For example, if RPV depressurization was determined to be successful in Level 1 PRA [e.g., no common cause failure (CCF) of the pressurizer SRVs to open], then that information should be transferred to Level 2 to preclude having common cause SRV failure be the cause of RPV depressurization failure if the same question is asked in the Level 2 PRA.	
L1-B5	As an example to support SR L1-B5 Capability Category I, if a small number of accident sequence end states (e.g., PDSs) are used, then the subsuming of the Level 1 accident sequences involves conservative modeling of the PDS characteristics to adequately cover the range of the different contributors to this small number of PDSs. "Conservative" in this context implies that accident sequences are grouped in a manner that skews the distribution of frequency among RCs toward those representing earlier and/or larger releases of fission products.	Examples of the factors that will affect the PDS variation are those given in SR L1-A1 and SR L1-A2 as well as those due to plant design or operational considerations.
L1-B6	See Commentary for SR L1-B5. With respect to conservative grouping, in cases where there is significant variability with respect to a particular attribute (e.g., the availability of a particular component), subsume the less favorable conditions (in terms of effect on radionuclide release magnitude and timing) within the group and transfer the information to the Level 2 PRA.	When grouping sequences within a PDS bin, consider the RC characteristics of the sequences as well as the frequency. PDSs should be assigned for sequences that have similar characteristics such that there should not be a large difference between sequences in a bin. Note that if PDSs are seen to have too wide a range of characteristics, additional bins may be needed to ensure that the variability between sequences is represented adequately.
L1-B7	Any simplifications made by the analyst should be identified to facilitate review. The degree of transfer that is "sufficient" should be able to be justified based on the intended application of the model.	The total CDF is transferred from the Level 1 PRA to the Level 2 PRA. However, the frequency of individual accident sequences with contributions below a particular threshold (1% of the total CDF, for example) can be allocated to a representative group (or groups) as a whole.
L1-B8	No commentary provided.	
L1-B9	No commentary provided.	
L1-B10	When using expert judgment, it is recommended that the analyst identify the basis for the parameter values such as pointing to completed experiments or computer analyses.	When using expert judgment, it is recommended that the analyst identify the basis for the parameter values such as pointing to completed experiments or computer analyses.

Table 2-A.2.1-4 Commentary to Supporting Requirements for HLR-L1-C

Index No. L1-C	Commentary
L1-C1	Note with respect to SR L1-C1*(a)*, the information required for the Level 2 PRA could be accident sequences and corresponding frequencies, dependencies, or system successes.
L1-C2	No commentary provided.

2-A.2.2 Commentary to Containment Performance Analysis (CP)

This Section presumes the existence of some type of passive structure surrounding the reactor with the capacity to withstand the conditions resulting from a design basis accident and retain a large portion of radioactive materials for beyond design basis accidents. The most common form of such a passive structure is a containment building (traditionally steel shell, steel-lined reinforced concrete, or steel-lined pre-stressed concrete), which often includes active and passive safeguard systems (e.g., distributed sprays, coolers, and passive pressure suppression devices). Where such a structure does not exist (e.g., a filtered confinement), portions of the analysis described in this section are not entirely applicable. Differences in requirements for pressure-retaining containment structures and filtered confinement structures are noted where appropriate.

It is expected that the requirements in the containment performance analysis would be examined for each POS considered in the Level 2 PRA, taking into account changes in containment configuration that can occur as operations shift from at-power conditions to refueling and the transition states in between. Care should therefore be taken to interpret the requirements in a manner that applies the expected changes in containment configuration. For example, during at-power operating conditions, the term "containment failure mechanism" is clearly understood in the context of structural failure or loss of isolation of an intact and isolated containment pressure vessel. This same term might necessarily represent the "normal" condition during certain periods of reactor shutdown. During shutdown, the "failure mechanism" of the containment pressure boundary (hatches, penetrations, etc.) could be treated as an assured condition due to the physical configuration of the containment. There might also be transition states in which (for example) hatches or penetrations might not be as completely secured as for at-power operation (e.g., some bolts removed from one or more flanges). This condition could be treated as a "degraded" initial state of the containment capacity. The SRs in the containment performance analysis should be applied for each of these conditions, as appropriate for the POS under evaluation.

This Section provides commentary for SRs contained in Table 2-2.2-2, Table 2-2.2-3, Table 2-2.2-4, Table 2-2.2-5, and Table 2-2.2-6 in Part 2 of this Standard. The following tables provide the commentary or additional material for the SRs helpful in understanding the intent of the requirement.

Table 2-A.2.2-1 Commentary to High Level Requirements for Containment Performance Analysis (CP)

Designator	Commentary
HLR-CP-A	No commentary provided.
HLR-CP-B	No commentary provided.
HLR-CP-C	No commentary provided.
HLR-CP-D	No commentary provided.
HLR-CP-E	No commentary provided.

Table 2-A.2.2-2 Commentary to Supporting Requirements for HLR-CP-A

Index No. CP-A	Commentary
CP-A1	Containment failure mechanisms should reflect the specific containment design. All containments will be expected to have overpressure failure and penetration failures. The purpose of the plant-specific investigation is to supplement a standardized list of "common" containment failure mechanisms from studies of other plants with similar containment design features, to the specific to the plant under consideration. Concrete containments will also be susceptible to liner tears, and steel containments may also be susceptible to thermally induced failures and vacuum induced compression failures. In identifying containment failure mechanisms, note that chemical composition of containment structures can impact aspects of the containment failure process such as basemat erosion and radiological degradation of cable insulation and/or penetration materials. SR L1-A1 should also identify those containment bypass and loss of isolation events that directly contribute to the core damage sequence. However, there are many failures of containment isolation states that are indeterminate from the perspective of a core damage sequence. Such failures can include potential containment failure mechanisms that result from loss of containment isolation, such as through open containment vents, failed containment penetrations, or cavity drains. Such containment failure modes should also be considered in identifying the complete set of containment failure mechanisms.
CP-A2	Failure of containment isolation due to the inability of containment isolation valves to close may be considered within the scope of the Level 1 PRA and should be transferred into the Level 2; alternatively, it may be developed as a new system in HLR-PT-C. These failures are not considered in the scope of SR CP-A2. Use of generic testing information may be used to identify applicable failure modes. Failure mechanisms should be evaluated for applicability to the plant being evaluated. Non-condensable gases may include air, containment atmosphere inerting gas, hydrogen, and other non-condensable gases generated by in-vessel oxidation of metallic core components or ex-vessel (e.g., core concrete interactions). Dynamic loads are related to effects such as BWR RPV blowdown through SRVs, downcomers, or other bypasses that impose extraordinary loads on the containment boundary or critical containment components. These loads can be exacerbated by containment water levels above design or by water temperatures above design. Some PWRs may have in-containment refueling water storage tanks (RWSTs) that may also lead to the consideration of containment challenges related to hydraulic loads. Note that containment overpressure from hydrogen combustion should include consideration of the point at which deflagration can transition to detonation [i.e., deflagration to detonation transition (DDT)].
CP-A3	Examples of these phenomena include material creep, or seal failure due to sustained exposure to high temperatures, and radiation damage to containment sealant materials. Severe accident conditions may impose high temperature and radiation challenges on seals and sealants/elastomers, which may impact integrity of containment penetrations. Chemical reaction sand radiological exposure may impact cable insulation. Chemical process may also impact basemat erosion, fission product release, and speciation.

Table 2-A.2.2-2 Commentary to Supporting Requirements for HLR-CP-A (Cont'd)

Index No. CP-A	Commentary
CP-A4	The included examples are not expected to be exhaustive. It is the responsibility of the analyst to review the containment design to confirm credible containment failure mechanisms have been adequately considered. Analysis of the hydrogen detonation event at Fukushima-Daichi suggests that it may be possible for detonations outside the containment to create conditions that could challenge its integrity. In addition to direct pressure loads, detonations outside containment could also cause motion of piping that penetrates the containment, which could challenge containment penetrations. The impact on containment integrity associated with a detonation outside the containment could be considered a source of uncertainty and can be identified and characterized in HLR-CP-D.
CP-A5	No commentary provided.
CP-A6	Sources can include generic industry experience regarding results of 10 CFR 50 Appendix J [2-A-10] testing on containment, penetrations, operational issues associated with open containment for maintenance/refueling, hatches left open, hatches or closures not tensioned to the correct torque, incorrect or deficient seal material in place, containment flaws, and containment corrosion leading to loss of capacity capability.
CP-A7	The analyst should look for potentially subtle pathways that could allow combustible and non-combustible gases to enter the auxiliary building during a severe accident that could compromise the ability of these buildings to retain fission products due to combustion or overpressurization. Pathways may involve overpressurization of piping or ducting that allows communications between the containment and the auxiliary building. These pathways can introduce combustible gases in areas where they may pose hazards to operating equipment and secondary barriers.
CP-A8	Note that the term "failure mechanisms" was used as this requires a defined location. For example, hatches fail at one location with a specified area. Some containment failures will be at the beltline, others at the basemat.
CP-A9	No commentary provided.

Table 2-A.2.2-3 Commentary to Supporting Requirements for HLR-CP-B

Index No. CP-B	Commentary
CP-B1	No commentary provided.
CP-B2	"Failure" could be defined as a maximum global membrane strain away from discontinuities of 1% for the assessment of ultimate pressure capacity for cylindrical reinforced concrete containments. One common approach to determining the capacity limit(s) for structural materials is to assume bounding constant temperature material properties as part of the method selected to evaluate the structural capacity of the containment or other buildings. Multiple capacity limits might be defined depending on the number and type of failure mechanisms included.

Table 2-A.2.2-3 Commentary to Supporting Requirements for HLR-CP-B (Cont'd)

Index No. CP-B	Commentary	
CP-B3	One option would be to provide an example where the analyst may apply a scalar to the containment design pressure based on observed ratios of failure-to-design pressure calculations for other similar containment structures. Another option would be to demonstrate that quantitative results of reference plant calculations are conservatively adapted to the plant and accident scenario of interest. Assuming an ultimate containment pressure of 2 times the containment design pressure is consistent with scaled failure tests and ultimate strength analyses of containments typically that show containment failure pressures to be more than a factor of 2 above the calculated containment design pressure. Selection of scaling factors greater than 2.0 should be carefully assessed as there is a concern that as the structure transitions further above its design pressure, non-linear factors and structural discontinuities (e.g., large penetrations) may become important factors in defining the ultimate containment strength. It is also expected that the analyst confirms that when using the reference test data, that the tests were conducted in a way that properly scaled the containment design features and the experiments were conducted on a containment design that is similar to the plant design or that adjustments used to address deviations are conservative. Quasi-static containment capability analyses may not be appropriate where high hydrogen concentrations exist. This is more likely for small volume containments or containments with geometries that support hydrogen pocketing or collection in small volume areas.	A validated model implies (1) use of a standard structural analysis code, such as ANSYS, that is capable of modeling geometry and material, (2) a structural model developed with standard modeling practice and (3) an independent review of the methodology and results. This is typical of standard engineering structural analysis. In many instances, containments are designed to standard conditions and their performance is validated by standard computer methods. When such analyses are available and the design analysis is directly applicable to the plant with perhaps minor differences, and provides the necessary information to meet this SR, such analyses should be considered sufficient to meet the intent of this SR. Existing design-specific structural analyses directly applicable to the plant specific containment design may be used, as available. Where available, it is acceptable to use results of applicable experimental measurements of containment performance for specific containment failure mechanisms. Quasi-static containment capability analyses may not be appropriate where high hydrogen concentrations exist. This is more likely for small volume containments or containments with geometries that support hydrogen pocketing or collection in small volume areas.
CP-B4	A discussion of methodology for including degraded conditions in a containment structural analysis and example analyses of various degraded (e.g., aged) containment designs can be found in NUREG/CR-6920 [2-A-11]. Known degradations should be considered in the failure analysis. Unknown degradations can be dealt with as part of the uncertainty characterization.	
CP-B5	Thermal-mechanical challenges are based on the results of severe accident analyses.	Thermal-mechanical challenges are based on the results of severe accident analyses.
CP-B6	Buildings outside containment structures may include BWR reactor building, PWR auxiliary building, turbine buildings, or other structures that may be impacted by the consequences of the severe accident.	No commentary provided.
CP-B7	No commentary provided.	No commentary provided.
CP-B8	No commentary provided.	No commentary provided.

Table 2-A.2.2-4 Commentary to Supporting Requirements for HLR-CP-C

Index No. CP-C	Commentary	
CP-C1	The time duration of the load would need to be accounted for in a creep analysis for the containment structure. Typically, creep failure modes in the time frame of a severe accident would require extremely high temperatures to have a significant impact on the containment failure characteristics. This is considered to be beyond the state of practice and is therefore not included in this SR. Expert judgment in concert with experimental or computational insights may be used to establish failure criteria/fragility curves.	The time duration of the load would need to be accounted for in a creep analysis for the containment structure. Typically, creep failure modes in the time frame of a severe accident would require extremely high temperatures to have a significant impact on the containment failure characteristics. This is considered to be beyond the state of practice and is therefore not included in this SR. Expert judgment in concert with experimental or computational insights may be used to establish failure criteria/fragility curves.
CP-C2	Plant operational modes refer to the operational status of the plant. These states include "full power operation," shutdown, and refueling. When treating shutdown modes, consider including transition modes within the shutdown state. Note that for some modes of reactor operation, containment failure limits and failure modes may differ from those "at-power" due to changes in containment closure requirements. Particular attention should be paid to justifying the use of "at-power" fragility curves for severe accident challenges should be representative for low power operation. However, use of "at power" fragility curves for shutdown modes of will be non-conservative for operating states allowing partial containment closure. Note that containment failure mechanisms for CC-I are expected to be conservatively biased, that is result in lower failure thresholds and/or more adverse failure modes. It is conservative to treat shutdown containment states as an open containment. Containment conditions for low power modes will be similar to "at power." When treating shutdown modes note that containment conditions are dependent on shutdown state and plant procedures. In some instances, containment may not be closed but not capable of withstanding full design pressure.	Plant operational modes refer to the operational status of the plant. These states include "full power operation," shutdown, and refueling. When treating shutdown modes, consider including transition modes within the shutdown state. Note that for some modes of reactor operation, containment failure limits and failure modes may differ from those "at-power" due to changes in containment closure requirements. Particular attention should be paid to justifying the use of "at-power" fragility curves for severe accident challenges should be representative for low power operation. However, use of "at power" fragility curves for shutdown modes of will be non-conservative for operating states allowing partial containment closure. Realistic analyses are performed using state-of-practice structural analysis methods and are evaluated using realistic material properties. Realistic analyses typically involve a finite element representation of the structural component being analyzed, realistic failure limits and may allow structural responses to extend into the plastic region. Bounding analyses are intended to be biased calculations that indicate margin to failure. Analyses that use elastic property failure limits and lower limit of material properties in a containment failure analysis would be considered bounding.
CP-C3	"Conservative" in this context implies that the location and size of failure would result in an earlier, larger, or both earlier and larger release of fission products to the environment than would result from a realistic analysis. In selecting location (or range of locations) the downstream application may be relevant. Note that for some modes of reactor operation, failure limits may differ from those used for at-power operation due to changes in containment closure requirements. Thus, using the "at-power" fragility curves for severe accident challenges during all modes of operation may not be appropriate (see commentary on SR CP-C2).	Realistic assessments of failure area are expected for the lower pressure failure (more likely) containment failure modes. For more robust containment SSCs (e.g., hatches at power), a less detailed analysis is reasonable and containment failure areas could be conservatively applied (i.e., larger areas). Note that for some modes of reactor operation, failure limits may differ from those used for at-power operation due to changes in containment closure requirements. Thus, using the "at-power" fragility curves for severe accident challenges during all modes of operation may not be appropriate (see commentary on SR CP-C2).

Table 2-A.2.2-4 Commentary to Supporting Requirements for HLR-CP-C (Cont'd)

Index No. CP-C	Commentary	
CP-C4	Generic fragility data may include fragility test data, generic seismic, or HWs qualification test data, earthquake experience data, and so on. Conservative estimation could include using bounding estimates to relate external hazards to containment failure mechanisms. For seismic events in particular, consider that relative motions between the auxiliary building (or equivalent) and containment can overstress containment penetrations and can cause localized failures. Large seismic events may also weaken or fail containment. Failure of isolation due to relay chatter or other causes should be identified in the Level 1/Level 2 PRA Interface or developed as a new system in HLR-PT-C. Other external hazards can be reviewed for applicable containment failure modes and their associated failure probabilities (e.g., missile damage from HWs, hydrodynamic and buoyancy loading from external floods, failure modes from other external hazards).	For seismic events in particular, consider that relative motions between the auxiliary building (or equivalent) and containment can overstress containment penetrations and can cause localized failures. Large seismic events may also weaken or fail containment. Failure of isolation due to relay chatter or other causes should be identified in the Level 1/Level 2 PRA Interface or developed as a new system in HLR-PT-C. Other external hazards can be reviewed for applicable containment failure modes and their associated failure probabilities (e.g., missile damage from HWs, hydrodynamic and buoyancy loading from external floods, failure modes from other external hazards).
CP-C5	Note that the list of limitations is not expected to be exhaustive, rather, it should identify those limitations that have the potential to change the interpretation of the containment performance analysis results.	

Table 2-A.2.2-5 Commentary to Supporting Requirements for HLR-CP-D

Index No. CP-D	Commentary	
CP-D1	Parameter uncertainty includes uncertainty associated with specifying material physical properties and dimensions. Modeling uncertainty is focused on assumed models and can include treatment of material deterioration mechanisms, treatment of dynamic loads, treatment of symmetry, flaw distributions, and so on.	
CP-D2	Note that for some modes of reactor operation, containment failure limits, failure modes, and associated uncertainties may differ from those "at-power" due to changes in containment closure requirements.	Note that for some modes of reactor operation, containment failure limits, failure modes, and associated uncertainties may differ from those "at-power" due to changes in containment closure requirements. Particular attention should be paid to calculating the impact of uncertainty using the "at-power" fragility curves for severe accident challenges for shutdown modes of operation. Use of "at-power" containment failure values and associated uncertainties for shutdown conditions may be non-conservative.

Table 2-A.2.2-5 Commentary to Supporting Requirements for HLR-CP-D (Cont'd)

Index No. CP-D	Commentary
CP-D3	Uncertainty specification includes the area of break (e.g., leak vs. catastrophic), location (e.g., beltline, dome, basemat, etc.), and probability of the various failure conditions. The predicted opening size variation may be illustrated by performing sensitivity studies on the structural analysis model parameters and failure assumptions used for estimating opening size. Uncertainties in the containment failure location may be based on multiple potential containment failure locations for direct or indirect containment failure mechanisms. Failure locations that release into buildings adjacent to the containment will have different release characteristics than releases into the ground or direct to atmosphere.
CP-D4	Parameter uncertainty includes uncertainty associated with specifying material physical properties and dimensions. Modeling uncertainty can include treatment of material deterioration mechanisms, treatment of dynamic loads, treatment of symmetry, flaw distributions, and so on. Note use of elastic or plastic deformation models for specifying containment failure limits includes the uncertainty associated with model selection and model parameters. Characterization may include use of sensitivity studies regarding impact of parameter uncertainty and analysis assumptions on containment failure modes and limits. Characterization can also include confidence intervals on calculated containment structure fragility curves. Epistemic impacts may be treated within the fragility confidence limits or separately.

Table 2-A.2.2-6 Commentary to Supporting Requirements for HLR-CP-E

Index No. CP-E	Commentary
CP-E1	Note that for SR CP-E1(*f*), examples of material or geometric degradation due to adverse environmental conditions could include corrosion, concrete decomposition, and so on.
CP-E2	No commentary provided.
CP-E3	No commentary provided.
CP-E4	No commentary provided.
CP-E5	No commentary provided.

2-A.2.3 Commentary to Severe Accident Progression Analysis (SA)

Special care should be exercised in defining "bounding" values for input parameters of models because a "conservative" assumption in one area often produces a non-conservative outcome in another area. The primary resources for obtaining this information are deterministic computer code calculations of specific severe accident sequences. The requirements outlined in this section primarily address the quality, technical rigor, and documentation of these calculations. Requirements concerning the use or application of results generated by these calculations in a probabilistic logic model are stated in the Probabilistic Treatment of Accident Progression and Source Terms.

This Section provides commentary for SRs contained in Table 2-2.3-2, Table 2-2.3-3, Table 2-2.3-4, Table 2-2.3-5, Table 2-2.3-6, and Table 2-2.3-7 of Part 2 of this Standard. The following tables provide the commentary or additional material for the SRs helpful in understanding the intent of the requirement.

Table 2-A.2.3-1 Commentary to High Level Requirements for Severe Accident Progression Analysis (SA)

Designator	Commentary
HLR-SA-A	No commentary provided.
HLR-SA-B	No commentary provided.
HLR-SA-C	No commentary provided.
HLR-SA-D	No commentary provided.
HLR-SA-E	No commentary provided.
HLR-SA-F	No commentary provided.