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)

TRIAL USE AND PILOT APPLICATION

Publication of this standard for trial use has been approved by The American Society of Mechanical Engineers and the American Nuclear Society. Distribution of this standard for trial use and comment shall not continue beyond 36 months from the date of publication, unless this period is extended by action of the Joint Committee on Nuclear Risk Management. It is expected that following this 36-month period, this draft standard, revised as necessary, will be submitted to the American National Standards Institute (ANSI) for approval as an American National Standard. A public review in accordance with established ANSI procedures is required at the end of the trial-use period and before a standard for trial use may be submitted to ANSI for approval as an American National Standard. This trial-use standard is not an American National Standard.

Comments and suggestions for revision should be submitted to:

Secretary, Joint Committee on Nuclear Risk Management
The American Society of Mechanical Engineers
Two Park Avenue
New York, NY 10016-5990

The American Society of Mechanical Engineers

ANS

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(This Foreword is not a part ASME/ANS RA-1.2-2014, "Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactor (LWRs)".)

FOREWORD

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

The Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs) was initiated by the ANS Risk Informed Standards Committee (RISC) in 2005 and is currently within the responsibility of the JCNRM Subcommittee on Standards Development. The Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs) was developed to provide requirements for the evaluation of containment performance and radiological releases to the environment. The radiological releases considered result from postulated accidents that cause fuel damage. The requirements of this standard apply to the evaluation of risk informed applications that use radionuclide release information or as input to the determination of inputs for Level 3 PRA evaluations (e.g., ex-plant consequences). This standard addresses sequences initiated by internal or external events during all modes of operation for operating and evolutionary commercial light water reactor (LWR) nuclear plants. This standard is used in conjunction with the ASME/ANS PRA Standard RA-Sa-2009. Specifically, the applicable requirements of the ASME/ANS PRA Standard RA-Sa-2009 are also applicable to those comparable parts of the Level 2 Analysis. In addition, the Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs) is structured to provide the requirements for all of the hazards defined in ASME/ANS PRA Standard RA-Sa-2009 and analyzed with a Level 1 PRA. The original draft of this standard was developed in 2011 and has undergone several revisions prior to the current ballot.

This standard sets forth the criteria for the technical adequacy of a Level 2 analysis to support risk-informed decisions for commercial nuclear power plants. Supporting requirements are provided for determining the chronology and physical processes governing core damage progression, containment response, and radiological release to the environment as part of PRAs and related analysis methodologies. This standard establishes the requirements to characterize the fission product release frequencies for various containment performance outcomes.

Significant input has been received from the JCNRM, specifically the JCNRM Subcommittee on Standards Development (SC-SD). In addition, an SC-SD consensus ballot readiness review team provided a valuable assessment of the proposed Level 2 PRA Standard prior to its submittal for ballot.

Publication for Trial Use

The technical requirements in this standard are based on source material from the existing ASME/ANS PRA standard ASME/ANS RA-Sa-2009 as well as the draft PRA standard under development by JCNRM for Level 3 PRA. Although RA-Sa-2009 was revised in 2013 in ASME/ANS RA-Sb-2013

(Addendum B), the changes in Addendum B are not fully addressed in this Level 2 PRA trial use standard. JCNRM has approved the use of draft ANS standards with a requirement to follow up with changes to reflect changes in the supporting standards. Such changes could necessitate a need for revisions to this standard. The use of source material from not-yet-approved PRA standards and several other considerations have shaped the decision to issue this standard for trial use. It is expected that changes that may be required to account for changes to the supporting standards will be accomplished as part of the effort to upgrade this trial-use standard to the requirements of the American National Standards Institute.

This standard is intended to be used together with other PRA standards that cover different aspects of PRA. Specifically, this standard is intended to be used directly with ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications." ASME/ANS RA-Sa-2009 includes Level 1 PRA and large early release frequency (LERF) for internal events at-power, external events, internal flood, and internal fire.

The Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs) cross references supporting requirements related to Systems, Data, Success Criteria, and Human Reliability Analysis to those technical elements of ASME/ANS RA-Sa-2009. This is consistent with the approach used in the LE element in Section 2-2.8 of ASME/ANS RA-Sa-2009 and in other sections of ASME/ANS RA-Sa-2009.

The format for this standard was developed in 2005 when no "standard" format was available. Therefore, it is not consistent with some other published PRA Standards regarding chapter numbers. Following Trial Use, the format of the section numbering will be reevaluated.

This standard is issued for Trial Use. Feedback is requested regarding the standard in all areas including the following general areas:

- Ease of use
- Clarity of technical supporting requirements (SR)
- Difficulty in the incorporation of interface requirements
- Difficulties in interpretation related to:
  - Different hazards
  - Different Plant Operating States
- Ability to evaluate significance when multiple release categories are involved
- Adequacy of references to PRA elements in other standards (e.g., Human Reliability, Systems, and Data)

Specific areas for which feedback is requested are:

- The availability of a realistic HRA technique to be used to satisfy SR PT-D2 for Capability Category II
- The minimum requirements for a peer review team (number of members, total study duration, total on-site presence) – Section 5.4.4
- A review of the ER HLR and SRs to ensure that the requirements are sufficiently clear and not duplicative.
- For SR L1-B2, is greater specification on the treatment of failure to run duration needed to assess the operation of mitigation equipment during accident progression?

PREPARATION OF TECHNICAL INQUIRIES TO THE JOINT COMMITTEE ON NUCLEAR RISK MANAGEMENT

INTRODUCTION

NOTE FOR TRIAL USE: The text of this section describes the technical inquiry process for approved standards. However, during the trial use period, users are encouraged to provide feedback, ask questions, and interact with the Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs) project team. Such feedback may be provided via the Secretary of the Joint Committee on Nuclear Risk Management, as noted below.

The ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM) will consider written requests for the interpretation and revision of risk management standards and the development of new requirements as dictated by technological development. JCNRM's activities in this latter regard are strictly limited to interpretations of the requirements or to the consideration of revisions to the requirements on the basis of new data or technology. As a matter of published policy, The American Society of Mechanical Engineers (ASME) does not "approve," "certify," "rate," or "endorse" any item, construction, proprietary device, or activity, and, accordingly, inquiries requiring such consideration will be returned. Moreover, ASME does not act as a consultant on specific engineering problems or on the general application or understanding of the standard's requirements. If, based on the inquiry information submitted, it is the opinion of the JCNRM that the inquirer should seek assistance, the inquiry will be returned with the recommendation that such assistance be obtained.

To be considered, inquiries will require sufficient information for JCNRM to fully understand the request.

INQUIRY FORMAT

Inquiries shall be limited strictly to interpretations of the requirements or to the consideration of revisions to the present requirements on the basis of new data or technology. Inquiries shall be submitted in the following format:

(a) *Scope.* The inquiry shall involve a single requirement or closely related requirements. An inquiry letter concerning unrelated subjects will be returned;
(b) *Background.* State the purpose of the inquiry, which would be either to obtain an interpretation of the standard's requirement or to propose consideration of a revision to the present requirements. Concisely provide the information needed for JCNRM's understanding of the inquiry (with sketches as necessary), being sure to include references to the applicable standard edition, addenda, part, appendix, paragraph, figure, or table;
(c) *Inquiry Structure.* The inquiry shall be stated in a condensed and precise question format, omitting superfluous background information and, where appropriate, composed in such a way that "yes" or "no" (perhaps with provisos) would be an acceptable reply. This inquiry statement should be technically and editorially correct;
(d) *Proposed Reply.* State what it is believed that the standard requires. If, in the inquirer's opinion, a revision to the standard is needed, recommended wording shall be provided;
(e) *Typewritten/Handwritten.* The inquiry shall be submitted in typewritten form; however, legible, handwritten inquiries will be considered;
(f) *Inquirer Information.* The inquiry shall include the name, telephone number, and mailing address of the inquirer;

(g) *Submission.* The inquiry shall be submitted to the following address: Secretary, Joint Committee on Nuclear Risk Management, The American Society of Mechanical Engineers, Two Park Avenue, New York, NY 10016-5990.

USER RESPONSIBILITY

Users of this standard are cautioned that they are responsible for all technical assumptions inherent in the use of PRA models, computer programs, and analysis performed to meet the requirements of this standard.

CORRESPONDENCE

Suggestions for improvements to the standard or inclusion of additional topics shall be sent to the following address: Secretary, Joint Committee on Nuclear Risk Management, The American Society of Mechanical Engineers, Two Park Avenue, New York, NY 10016-5990.

COMMITTEE ROSTERS

CONTRIBUTORS TO THE SEVERE ACCIDENT PROGRESSION AND RADIOLOGICAL RELEASE (LEVEL 2) PRA STANDARD FOR NUCLEAR POWER PLANT APPLICATIONS FOR LIGHT WATER REACTORS (LWRs)

(The following is a roster of the Joint Committee on Nuclear Risk Management at the time of the approval of this standard.)

This standard was processed and approved for release for trial use and pilot application by the ANS/ASME Joint Committee on Nuclear Risk Management in accordance with procedures approved by ASME and ANS. At the time it approved this standard, the JCNRM had the following members:

ASME/ANS Joint Committee on Nuclear Risk Management (JCNRM)

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ASME/ANS RA-S-1.2-2014 (formerly ANS/ASME-58.24 of the Standards Committee of the American Nuclear Society) was responsible for development of this standard. It had the following membership when first formulated:

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1. INTRODUCTION

1.1 Objectives

This standard[1] sets forth the requirements for probabilistic risk assessments (PRAs) used to support risk-informed decisions for commercial light water reactor (LWR) nuclear power plants[2]. Unique requirements are specified as needed for specific reactor designs.

Sections 1, 3, and 5 of this standard generally mirror the analogous information in the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard, ASME/ANS RA-Sa-2009 [1][3]. In cases where deviations from that document are believed to be of particular interest, these deviations are underlined.

1.2 Coordination with Other Probabilistic Risk Assessment Standards

This standard is intended to be used together with other PRA standards that cover different aspects of PRA scope [1].

1.2.1 Interface with ASME/ANS RA-Sa-2009 and Other Level 1 PRA Standards

This standard is intended to be used directly with the PRA standard developed by the ASME and ANS, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME/ANS RA-Sa-2009[4] [1]. ASME/ANS RA-Sa-2009 [1] covers internal events and external hazards that might occur while the nuclear power plant is at-power [5].

1.2.2 Interface with a Level 3 PRA

The end point of a Level 2 analysis is the distribution of the core damage frequency (CDF) into a set of radionuclide release categories (RCs). These RCs 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).

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-Sa-2009 [1]. Rather, this standard supplements and extends the LERF portion of the ASME/ANS RA-Sa-2009 to include a more quantitative and comprehensive analysis of the full spectrum

1 The current standard, ASME/ANS RA-S-1.2-2014, is herein referred to as "this standard."

2 As currently written, this standard applies only to postulated accident sequences in commercial LWRs (currently operating nuclear plants and so-called evolutionary or advanced LWRs with sufficiently detailed design information to evaluate plant response to accident sequences involving substantial core damage). As noted in Section 1.3, revisions may be necessary so that it can be applied to next generation designs.

3 Numbers in brackets refer to corresponding numbers in Section 6, "References."

4 The ASME/ANS PRA Standard is herein referred to as "ASME/ANS RA-Sa-2009 [1]."

5 Another standard is being developed to address core damage accidents during low power/shutdown (LPSD) conditions. As a group, these standards provide the guidance for assessing the technical adequacy of Level 1 PRA analyses used to support risk-informed applications.

of possible radionuclide releases resulting from postulated severe accidents. The Level 2 PRA analysis provides a means of distributing the CDF into a set of RCs spanning the entire range of fission product release characteristics.

A subset of the RCs represent large early releases, which have the potential for significant offsite early health effects. ASME/ANS RA-Sa-2009 [1] includes requirements for estimating the frequency of large early releases as a metric for many PRA applications. Performing a full Level 2 analysis provides an opportunity for a refined determination of the LERF as a result of the greater degree of modeling detail compared to that typical of a LERF evaluation, as prescribed in the ASME/ANS RA-Sa-2009 Standard.

The LERF technical element of the ASME/ANS RA-Sa-2009 [1] PRA Standard remains as the appropriate reference for PRA applications that would need LERF for any or all Capability Categories.

This standard has added requirements for the evaluation of risk metrics other than solely LERF. These risk metrics primarily consist of other RCs in addition to LERF. This standard is also more explicit in preparing an interface with potential future use with a Level 3 PRA.

The completion of a Level 2 PRA according to this standard would meet the LERF requirement for each comparable Capability Category.

The completion of a LERF analysis according to the ASME/ANS RA-Sa-2009 [1] would also meet the LERF requirements for each Capability Category.

1.3 Scope

The scope of a PRA covered by 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 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 postulated accident sequences initiated from all modes of reactor operation (at-power, shutdown, and transition states). It also includes accident sequences initiated by internal events and/or external hazards addressed in ASME/ANS RA-Sa-2009 [1].

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-Sa-2009 [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 the occurrence at Fukushima Daiichi during March, 2011.

The requirements described in this standard address commercial LWRs (currently operating nuclear plants and so-called evolutionary or advanced LWRs with sufficiently detailed design information to evaluate plant response to accident sequences involving substantial core damage). Revisions may be necessary so that it can be applied to next generation designs. This standard is applicable throughout the life cycle of a plant. Of course, this applicability must recognize that some supporting requirements (SRs) cannot be met during the early phases of design and operation when data procedures, training, etc. are not available for evaluation.

The applicability to other LWR designs would have to be evaluated on a case-by-case basis. Caution must be exercised when applying these requirements to reactor and containment designs that are substantially different from operating LWR designs or current evolutionary LWR designs.

1.4 PRA Capability Categories

This standard is intended to support a wide range of risk-informed applications that require a corresponding range of PRA capabilities. 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 risk significance of the subject of the decision. In developing the different portions of the PRA model, it is recognized that not every item (e.g., system models) will be or need to be developed to the same level of detail, the same degree of plant-specificity, or the same degree of realism.

Although the range of capabilities required for each portion of the PRA to support an application falls on a continuum, three levels are defined and labeled either Capability Category I, II, or III so that requirements can be developed and presented in a manageable way. For three principal attributes of PRA, Table 1.4-1 describes the bases for defining the Capability Categories. This table was used to develop the SRs for each high-level requirement (HLR).

The intent of the delineation of the Capability Categories within the SRs is generally that the degree of scope and level of detail, the degree of plant-specificity, and the degree of realism increase from Capability Category I to Capability Category III. However, the Capability Categories are not based on the level of conservatism (i.e., the tendency to overestimate risk due to simplifications in the PRA) in a particular aspect of the analysis. 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. 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.

The bases for the PRA Capability Category assignments for this standard (Table 1.4-1) have been modified relative to the equivalent table, Table 1-1.3-2 of the ASME/ANS RA-Sa-2009 [1]. These changes are required because:

Attribute	Reason for Differences
Scope and Level of Detail	Scope expanded to recognize all modes of operation, initiating events, and mechanisms of containment failure and treatment of recovery after core damage. Clarification to identify that the resolution is directed at significant accident progression sequences (CC II) and all accident progression sequences (CC III).
Plant Specificity	Clarification to identify that the resolution is directed at significant accident progression sequences (CC II) and all accident progression sequences (CC III).
Realism	The distinctions with regard to realism are made to emphasize that the Level 2 PRA has many plausible outcomes, each of which are addressed to some degree of fidelity. A conservative treatment of parameters or models in Level 2 analysis affecting one outcome often results in a non-conservative (or at least an unrealistic) treatment of alternative outcomes. This Level 2 PRA treatment contrasts with the Level 1 PRA that is predominantly concerned with one outcome (core damage frequency). For example, a conservative treatment of some severe accident phenomena or characteristics of system performance in Level 2 PRA may increase the contributions (e.g., the frequency) of certain sequences and associated release categories but will necessarily decrease the contributions (non-realistic bias) to other sequences and release categories. In particular, in contrast to the ASME/ANS RA-Sa-2009 [1], the Level 2 PRA requirements described here address more end states than LERF. The definition of conservative can sometimes be counterintuitive. For example, under some circumstances, a change in a Level 2 PRA outcome can increase some consequence metrics while decreasing others. Thus, the meaning of conservative can depend on the situation and on the metrics of interest. Consequently, the manner in which realism is treated across the full spectrum of end states is a key aspect in the analysis of accident progression sequences and distinguishes among the assigned Capability Categories.

The boundaries between these Capability Categories can only be defined 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 technical elements or portions of the PRA within each element will not necessarily all fall within the same Capability Category, but rather will be distributed among all three Capability Categories. There also may be PRA elements or portions of the PRA within the elements that fail to meet the SRs for any 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).

Table 1.4-1 Bases for Level 2 PRA Capability Categories

Attributes of the PRA	Capability Category I	Capability Category II	Capability Category III
1. Scope and Level of Detail: The degree to which the scope and level of detail of the analysis are sufficient to capture the important physical phenomena relevant to the plant design.	Resolution and specificity 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 [see Note (1)].	Resolution and specificity sufficient to identify the operating modes, initiating events, system failures, system operating characteristics, mechanisms of containment failure, and severe accident progression phenomena that contribute to *significant* accident progression sequences.	Resolution and specificity sufficient to identify the operating modes, initiating events, system failures, system operating characteristics, mechanisms of containment failure, and severe accident progression phenomena that contribute to *all* accident progression sequences.
2. Plant-specificity: The degree to which plant-specific information is incorporated such that the as-built and as-operated plant is addressed.	Use of *generic* data/ models is acceptable except for the need to account for the unique design and operational features of the plant.	Use of *plant-specific* data/models for evaluating challenges to containment integrity and fission product release characteristics for *significant* accident progression sequences.	Use of *plant-specific* data/models for evaluating challenges to containment integrity and fission product release characteristics for *all* accident progression sequences.
3. Realism: The degree to which realism is incorporated such that the expected responses of the plant and containment are addressed.	*Bounding or conservative* characterization of the frequency and physical characteristics (magnitude, timing, etc.) of radiological releases for accident progression sequences generated in the Level 2 PRA.	*Realistic* characterization of the frequency and physical characteristics (magnitude, timing, etc.) of radiological releases for *significant* progression accident sequences generated in the Level 2 PRA.	*Realistic* characterization of the frequency and physical characteristics (magnitude, timing, etc.) of radiological releases for *all* accident progression sequences generated in the Level 2 PRA.

NOTES:

(1) In this context, "unmitigated system failures" refers to failures of active or passive systems (including building structures) that are not restored or mitigated after the onset of core damage by, for example, human actions directed by severe accident management guidelines (SAMGs).

1.5 Requirements for the PRA Elements

The technical requirements for each PRA technical element are defined in Sections 4.2 through 4.8 of this standard. The following paragraphs provide an overview of the requirements and some guidance on their interpretation.

This standard specifies technical requirements for the PRA elements listed in Table 1.5-1.

1.5.1 High-Level Requirements

A set of objectives and HLRs is provided for each PRA technical element in the Technical Requirements in Section 4 of this standard. The HLRs set forth the minimum requirements for a technically acceptable baseline PRA independent of the application. The HLRs are defined in general terms and present the top-level logic for the derivation of more detailed SRs. The HLRs reflect not only the diversity of approaches that have been used to develop the existing PRAs, but also the need to accommodate future technological innovations.

1.5.2 Supporting Requirements

The SRs for the technical elements are presented as action statements in the Technical Requirements in Section 4 of this standard using the three Capability Categories. The SRs are numbered and labeled to identify the HLR that is supported. For each Capability Category, the SRs define the minimum requirements necessary to meet that Capability Category. In these tables, some action statements apply to only one Capability Category, while some extend across two or three Capability Categories. When an action spans multiple Capability Categories, it applies equally to each Capability Category. When necessary, the differentiation between Capability Categories is made in other associated SRs. The interpretation of a SR whose action statement spans multiple Capability Categories is stated in Table 1.5-2. It is intended that by meeting all the SRs under a given HLR, a PRA will meet that HLR. The Technical Requirements section of each part of this standard also specifies the required documentation to facilitate PRA applications, upgrades, and peer review.

The SRs specify what to do rather than how to do it, and in that sense, specific methods for satisfying the requirements are not prescribed. Nevertheless, certain established methods were contemplated during the development of these requirements, for example, the use of codes such as MELCOR or Modular Accident Analysis Program (MAAP), which are state-of-the-art codes and widely accepted computational tools for severe accident analysis when applied within their established domain of applicability (see Table 4.4-3), although the use of other codes may also be acceptable. Alternative methods and approaches to satisfy the requirements of this standard may be used if they meet the HLRs and SRs presented in this standard. The use of any particular method for meeting an SR shall be documented and shall be subject to review by the peer review process described in Section 5.

All Notes and Commentaries that follow many SRs are non-mandatory.

Table 1.5-1 PRA Technical Elements Addressed by Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs)

Hazard Type	Hazard Group	Technical Elements
All Hazard Types	All Hazard Groups	Level 1/Level 2 PRA Interface -- Accident Sequence Grouping (L1) Containment Capacity Analysis (CP) Severe Accident Progression Analysis (SA) Probabilistic Treatment of Event Progression and Source Terms (PT) Radiological Source Term Analysis (ST) Evaluation and Presentation of Results (ER) Interface Between Level 2 PRA and Level 3 PRA (L3)

Table 1.5-2 Interpretation of Supporting Requirements

Action Statement Spans	Peer Review Finding	Interpretation of the Supporting Requirements
All three Capability Categories (I/II/III)	Meets SR	Capable of supporting applications in all Capability Categories
	Does not meet SR	Does not meet the minimum standard
Single Capability Category (I, II, or III)	Meets individual SR	Capable of supporting applications requiring that Capability Category or lower
	Does not meet SR	Does not meet the minimum standard
Lower Two Capability Categories (I/II)	Meets SR for CC I/II	Capable of supporting applications requiring Capability Category I or II
	Meets SR for CC III	Capable of supporting applications in all Capability Categories
	Does not meet SR	Does not meet the minimum standard
Upper Two Capability Categories (II/III)	Meets SR for CC II/III	Capable of supporting applications in all Capability Categories
	Meets SR for CC I	Capable of supporting applications requiring Capability Category I
	Does not meet SR	Does not meet the minimum standard

1.6 Risk Assessment Application Process

The use of a PRA and the Capability Categories that are needed for each of the PRA technical elements will differ among applications. PRA technical adequacy is assessed for applicable parts of a PRA and each associated SR rather than by specifying a Capability Category for the whole PRA. Therefore, only those parts of the PRA required to support the application in question need the Capability Category appropriate for that application. For a given application, supplementary analyses may be used in place of or to augment those aspects of a PRA that do not fully meet the requirements in the Technical Requirements section of this standard. Requirements for supplementary analysis are outside the scope of this standard.

Section 1-3 of ASME/ANS RA-Sa-2009 [1] describes a five-stage process for determining the PRA capabilities needed to support a particular application. That process is summarized below.

Stage A: Define the application in terms of the structures, systems, and components (SSCs) and activities affected by the proposed change. Determine the portions of the PRA affected by the application, the hazard group(s) needed to be addressed in the application, the scope within the PRA related to the application, and the risk metrics needed to support the application (refer to the Stage B description for possible considerations associated with Level 2 analysis).

Stage B: Evaluate the relevant portions of the PRA to determine whether its scope and level of detail are sufficient for the application. If relevant portions of the PRA are found to be lacking in one or more areas, determine the upgrades or supplementary analyses needed. As part of evaluating the relevant portions of the PRA to determine the sufficient scope to support the application, it is expected that the determination would be made regarding whether the ASME/ANS RA-Sa-2009 [1] technical element LE is sufficient or whether this standard's requirements would be appropriate. This evaluation would also include the assessment regarding the needed risk metrics (e.g., offsite consequence evaluation).

Stage C: Determine whether the capability requirements for the SRs from the relevant portions of the standard are sufficient to support the application. If not, the SRs may be augmented with supplementary requirements as described in Stage E.

Stage D: Compare each relevant portion of the PRA to the appropriate SRs to determine whether the PRA has adequate technical capability, needs upgrading to meet the appropriate SRs, or needs supplementary analyses as described in Stage E.

Stage E: The relevant portions of the PRA, upgraded or supported by supplementary analyses if necessary, are used to support the application. This activity is outside the scope of this standard, as are the criteria for judging the quality of any supplementary analyses performed to support the application.

For more detail regarding this process, the reader is referred to ASME/ANS RA-Sa-2009 [1].

1.7 Risk Assessment Technical Requirements

1.7.1 Purpose

The purpose of this section is to provide requirements by which adequate PRA capability can be identified when a PRA is used to support applications of risk-informed decision-making. This section also includes general requirements for process checking of analyses and calculations and for use of expert judgment.

1.7.2 Process Check

Analyses and/or calculations used directly by the PRA (e.g., calculations of probability distributions) or used to support the PRA (e.g., severe accident progression or source term calculations) shall be reviewed by knowledgeable individuals who did not perform those analyses or calculations. Documentation of this review may take the form of hand-written comments, signatures, or initials on the analyses/calculations, formal sign-offs, or other equivalent methods.

1.7.3 Use of Expert Judgment

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

Guidance from NUREG-2117 [2], NUREG/CR-6372 [3], and NUREG-1563 [4] may be used to meet the requirements in this paragraph. Other approaches or a mix of these may also be used.

EXAMPLES: Use of expert judgment to resolve difficult issues includes Pacific Gas and Electric's Diablo Canyon seismic study [5] and the Yucca Mountain project's study of volcanic hazards [6]. These reports provide useful insights into both the strengths and the potential pitfalls of using experts. A review of expert aggregation methods, the different types of consensus, and issues with resolving disagreements among experts can be found in Appendix J of NUREG/CR-6372 [3].

1.7.3.1 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).

1.7.3.2 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).

1.7.3.3 Determination of the Need for Outside Expert Judgment

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

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

(a) complex experimental data exist that the analysts know have been interpreted differently by different outside experts
(b) more than one conceptual model exists for interpreting the technical issue, and judgment is needed as to the applicability of the different models

(c) judgments are required to assess whether bounding assumptions or calculations are appropriately conservative
(d) uncertainties are large and significant, and judgments of outside technical experts are useful in illuminating the specific issue

1.7.3.4 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 will use a panel of expert evaluators and a facilitator/integrator

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

1.7.3.5 Identification and Selection of Evaluator Experts

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

(a) their own input
(b) their representation of the community distribution

1.7.3.6 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

1.7.3.7 Responsibility for the Expert Judgment

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

1.7.4 Derivation of PRA Requirements

Objectives were established for each technical element used to characterize the respective scope of a PRA. The objectives reflect substantial experience accumulated with PRA development and usage and are consistent with the PRA Procedures Guide [7], NEI 00-02 [8], and NEI 05-04 [9] Peer Review Process Guidance, where applicable. These objectives form the basis for the development of the HLRs for each technical element that were used, in turn, to define the SRs.

In setting the HLRs for each technical element, the goal was to derive, based on the objectives, an irreducible set of firm requirements applicable to PRAs that support all levels of application to guide the development of SRs. This goal reflects the diversity of approaches that have been used to develop existing PRAs and the need to allow for technological innovations in the future. An additional goal was to derive a reasonably small set of HLRs that capture all the important technical issues that were identified in the efforts to develop this standard and to develop and implement a peer review guidance process for this standard that follows the approach and intent of NEI 00-02 [8] and NEI 05-04 [9] PRA Peer Review Process Guidance (for internal event PRAs).

The HLRs generally address attributes of the PRA element such as:

(a) scope and level of detail
(b) model fidelity and realism
(c) output or quantitative results (if applicable)
(d) documentation

Three sets of SRs were developed to support the HLRs in the form of action statements for the various capability categories in the standard. Therefore, a complete set of SRs is provided for each of the three PRA Capability Categories described in Section 1.4.

1.7.5 PRA Requirements

Tables of HLRs and SRs for the technical elements are provided for each PRA scope. The SRs are numbered and labeled to identify the HLR that is supported. For each Capability Category, the SRs define the minimum requirements necessary to meet that Capability Category. In these tables, some action statements apply to only one Capability Category, and some extend across two or three Capability Categories. When an action spans multiple Capability Categories, it applies equally to each Capability Category. When necessary, the differentiation between Capability Categories is made in other associated SRs. The interpretation of a Supporting Requirement whose action statement spans multiple Capability Categories is stated in Table 1.5-2. It should be noted that some action statements span Capability Categories II and III because the authors were unable to specify a distinguishing requirement for Capability Category III at this time. It is intended that by meeting all the SRs under a given HLR, a PRA will meet that HLR.

1.8 PRA Configuration Control

Section 3 provides requirements for configuration control of a PRA (i.e., maintaining and upgrading a plant-specific PRA) such that the PRA reflects the as built, as-operated facility to a degree sufficient to support the application for which it is used.

1.9 Peer Review

Section 5 provides the general requirements for a peer review to determine if the PRA methodology and its implementation meet the requirements of the Technical Requirements section of this standard. SRs on documentation to facilitate peer review are found in Table 4.7-2 of this standard.

1.10 Addressing Multiple Hazards

The technical requirements to determine the technical adequacy of a Level 1 PRA for different hazard groups to support applications are presented in Parts 2 through 10 of ASME/ANS RA-Sa-2009 [1].

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 by the state of the art. For example, there are uncertainties that are unique to the modeling of 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. These screening approaches are unique to each hazard group. For many applications, it is necessary to consider the combined impact on risk from those hazard groups for which it cannot be demonstrated that the impact on the decision being made is insignificant. This can be done by using a single 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. To support this objective, the standard is structured so that requirements for the analysis of the PRA results including the identification of significant contributors, the identification and characterization of sources of uncertainty, and the identification of assumptions are included separately in each part.

In some cases, the requirements for developing a PRA model in Parts 3 through 10 of ASME/ANS RA-Sa-2009 [1] refer back to the requirements of Part 2 of ASME/ANS RA-Sa-2009. The requirements of Part 2 of ASME/ANS RA-Sa-2009 should be applied to the extent needed given the context of the modeling of each hazard group. In each Part, many of the requirements that differentiate between Capability Categories either directly or by incorporating the requirements of Part 2 [1] do so on the basis of the treatment of significant contributors and significant accident sequences/cut sets for the hazard group being addressed. Because, as discussed above, there are differences in the way the PRA models for each specific hazard group are developed, the requirements are best treated as being self-contained for each hazard group separately when determining significant contributors and significant accident sequences/cut sets. In other words, for the Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for Light Water Reactors (LWRs), this significance is identified separately with respect to the CDF, LERF, and other RCs for each hazard group. While there is a need in some applications to assess the significance with respect to the total CDF, LERF, or radionuclide release, this assessment has to be done with a full understanding of the differences in conservatism and level of detail introduced by the modeling approaches for the different hazard groups as well as within each hazard group.

To determine the Capability Category at which the SRs have been met, it is necessary to have a definition of the term “significant.” Consequently, the term “significant” is used with various definitions in this standard and is thereby explicitly incorporated into specific SRs. Generally, the philosophy used in Capability Category II ensures a higher level of realism for significant contributors. This manifests itself in SRs related to the scope of plant-specific data, detailed human reliability analysis (HRA; versus screening values), common cause failure (CCF) treatment, documentation, and others.

The only consequence of not meeting the standard definition of significant for a specific SR is that the PRA would not meet Capability Category II for that SR. Thus, in the context of an application, if a hazard group is a small contributor, it should be acceptable to meet Capability Category I by using screening human error probabilities (HEPs), not by using plant-specific data for equipment reliability, etc. The applicable portion of the PRA will simply be considered as meeting Capability Category I for that specific SR of that hazard group.

Additionally, from a practical standpoint, PRA models are generally developed on a hazard group basis (i.e., a fire PRA, a seismic PRA, a high wind PRA, etc.). While they may be integrated into a single model with multiple hazards, the development is done on a hazard group basis. In Capability Category II, this standard strives to ensure that the more significant contributors to each hazard group are understood and treated with an equivalent level of resolution, plant specificity, and realism, so as to not skew the results for that hazard group. The definitions also acknowledge that there may be cases where the proposed quantitative definition is inappropriate (e.g., the hazard group risk is very low or bounding methods are used).

To summarize, the definitions that use the term "significant" simply help to define how much realism is necessary to meet Capability Category II of some SRs. They are NOT intended to be definitions of what is significant in a particular application. 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 the Risk Assessment Application Process (see Section 1.6).

2. ACRONYMS AND DEFINITIONS

2.1 Acronyms

ABWR: Advanced Boiling Water Reactor
AC: Alternating Current
ANS: American Nuclear Society
ASME: American Society of Mechanical Engineers
BWR: Boiling Water Reactor
BWST: Borated Water Storage Tank
CCF: Common Cause Failure
CDF: Core Damage Frequency
CET: Containment Event Tree
CST: Condensate Storage Tank
DW: Drywell
EOP: Emergency Operating Procedure
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
ISLOCA: Interfacing Systems Loss of Coolant Accident
LERF: Large Early Release Frequency
LOCA: Loss of Coolant Accident
LOOP: Loss of Offsite Power
LOSP: Loss of Offsite Power
LPSD: Low Power and Shutdown
LRF: Large Release Frequency
LWR: Light Water Reactor
MAAP: Modular Accident Analysis Program
MOV: Motor-Operated Valve
NEI: Nuclear Energy Institute
NRC: Nuclear Regulatory Commission
NSSS: Nuclear Steam Supply System
PDS: Plant Damage State

POS: Plant Operating State

PRA: Probabilistic Risk Assessment

PSA: Probabilistic Safety Assessment

PWR: Pressurized Water Reactor

RC: Release Category or Radionuclide Release Category

RCS: Reactor Coolant System

RG: Regulatory Guide (an NRC issued communication)

RPV: Reactor Pressure Vessel

RWST: Refueling Water Storage Tank

SAMG: Severe Accident Management Guideline

SGTR: Steam Generator Tube Rupture

SR: Supporting Requirement

SRV: Safety Relief Valve

SSCs: Structures, Systems, and Components

THERP: Technique for Human Error Rate Prediction

2.2 Definitions

accident progression signature: See *signature*

accident progression event tree: See *containment event tree*

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. Examples of the framework may include the following: (a) only containment event trees (CETs; or the equivalent); (b) the CETs and bridge trees; (c) expanded Level 1 event trees and CETs; or (d) a single fully integrated model incorporating the equivalent of Level 1 event trees and CETs.

accident progression sequence: A unique combination of events that clearly delineates 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 (such as system, function, or operator performance) that can lead to undesired consequences, with a specified end state (e.g., core damage or large early release).

accident sequence, significant: See *significant accident sequence*

aleatory uncertainty: The uncertainty inherent in a nondeterministic (stochastic, random) phenomenon. Aleatory uncertainty is reflected 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.")

assumption: A decision or judgment that is made in the development of the probabilistic risk assessment (PRA) model. An assumption is either related to a source of model uncertainty or is related to scope or level of detail. An assumption related to a model uncertainty is made with the knowledge that a different reasonable alternative assumption exists. A reasonable alternative assumption is one that has broad acceptance within the technical community and for which the technical basis for consideration is at least as sound as that of the assumption being made. An assumption related to scope or level of detail is one that is made for modeling convenience. An assumption is labeled ''key'' when it may influence (i.e., have the potential to change) the decision being made. Therefore, a key assumption is identified in the context of an application.

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

availability: The complement of unavailability.

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

benevolent failure: A failure of an active or passive system component or a structural member of the reactor or containment pressure boundary that alters accident progression in a manner that reduces the severity of the current reactor or containment status or mitigates the consequences of subsequent events. An example is the failure of a safety/relief valve to reclose on demand, causing unintentional depressurization of the reactor coolant system (RCS). This event has the beneficial effect of reducing reactor vessel pressure, thereby reducing the potential for adverse creep rupture of the reactor coolant system (e.g., induced steam generator tube rupture (SGTR)) and (later in time) high-pressure failure of the reactor pressure vessel (RPV) lower head. Such failures are often precluded from consideration in the Level 1 PRA, but can be credited in the Level 2 analysis to facilitate a more realistic assessment of severe accident progression, especially when there is a clear link to severe accident conditions causing the failure.

bounding analysis: Analysis that uses assumptions such that the assessed outcome will meet or exceed the maximum severity of all credible outcomes.

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.

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

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

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

containment capacity analysis: Deterministic analysis of a containment structure to determine its capacity (or capability) to withstand defined internal or external loads or a specific challenge to its integrity.

containment bypass: A direct or indirect flow path that would allow the release of radioactive material to be transported directly to the environment without benefit of any attenuation in the containment.

containment challenge: Severe accident conditions (e.g., plant thermal hydraulic conditions or phenomena) that may compromise containment integrity, i.e., lead to containment failure. These conditions or phenomena can be compared with containment capacity to determine whether a containment failure mode results.

containment event tree (CET): A logic diagram that begins with a Level 1 PRA end state (e.g., accident sequence or plant damage) and progresses through a series of branches that: (1) represent expected system or operator performance that either succeeds or fails; (2) delineate the chronological and physical progression of core damage; (3) characterize containment response; and (4) represent processes affecting fission product release to the environment. The end states of the CET can be associated with release categories.

containment failure: Loss of integrity of the containment pressure boundary that has the potential for a release of radionuclides to the environment of sufficient magnitude to impact the application of interest.

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

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

core damage: Uncovery and heatup 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 offsite public health effects.

core damage frequency (CDF): Expected number of core damage events per unit of time.

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

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. (Epistemic uncertainty is sometimes also called ‘‘modeling uncertainty.’’)

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, etc.), and instrumentation and indication components (e.g., status indicator lights, meters, strip chart recorders, sensors, etc.). Equipment, as used in this standard, *excludes* electrical cables.

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

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

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

expert 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 event: An event 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. Events such as earthquakes, tornadoes, and floods from sources outside the plant are considered external events. (See also internal event). By historical convention, LOOP not caused by another external event is considered to be an internal event.

failure mechanism: Any of the processes that results 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 upon 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: The 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[6]. The fragility model used in seismic PRA is known as a double lognormal model with three parameters, which are the median acceleration capacity, the logarithmic standard deviation of the aleatory (randomness) uncertainty in capacity, and the logarithmic standard deviation of the epistemic (modeling and data) uncertainty in the median capacity.

[6] This is the input used in Level 2 PRA standard in association with the term "fragility."

hazard: An event or a natural phenomenon that poses some risk to a facility. Internal hazards include events such as equipment failures, human failures, and flooding and fires internal to the plant. External hazards include events such as flooding and fires external to the plant, tornadoes, earthquakes, and aircraft crashes.

hazard group: A group of similar hazards that are assessed in a PRA using a common approach, methods, and likelihood data for characterizing the effect on the plant. Typical hazard groups considered in a nuclear power plant PRA include: internal events, internal floods, seismic events, internal fires, high winds, external flooding, etc.

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

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

human reliability analysis (HRA): A structured approach used to identify potential human failure events and to systematically estimate the probability of those events using data, models, or expert judgment.

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

initiating event: A perturbation to 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 high winds), or combinations thereof.

initiator: See *initiating event*.

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

interfacing systems LOCA (ISLOCA): 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 over-pressurization of a low-pressure system when subjected to RCS pressure and can result in containment bypass.

internal event: A hazard group that encompasses events that result from or involve mechanical, electrical, structural, or human failures from causes originating within a nuclear power plant that directly or indirectly cause an initiating event and may cause safety system failures or operator errors that may lead to core damage. By historical convention, loss of offsite power, which may result from causes within or outside the plant, is considered an internal event (except when the loss is caused by another evaluated hazard group). Also by historical convention, internal flood and internal fire are separate hazard groups and thus not considered internal events.

large early release: The rapid, unmitigated release of airborne fission products from the containment to the environment 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 (LERF): Expected number of large early releases per unit of time.

large release frequency (LRF): Expected to be defined on a plant-specific, application-specific basis. This is to be defined by the users of the Level 2 PRA Standard. Typical definitions of "large release" that have been used previously are summarized in NUREG-2122 [10] and are further discussed in SECY-13-0029 [11].

level 1 analysis: Identification and quantification of the sequences of events leading to the onset of core damage.

level 1 PRA: See *level 1 analysis*.

level 2 analysis: Evaluation of reactor and containment response to accident sequences following physical damage to reactor fuel. The evaluation includes fuel rod damage and the release of radioactive fission products into the RCS and (possibly) to the containment and environment. This analysis starts with the onset of core damage and covers accident progression to the point of radioactive release to environment.

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

level 3 analysis: Estimation of the consequences of the release to the environment from radioactive materials, as identified in the Level 1/2 analyses.

level 3 PRA: A PRA that encompasses the Level 1, Level 2, and Level 3 analyses. See also *level 3 analysis*.

low power and shutdown (LPSD): Refers to modes of reactor operation at low power when subcritical, or when the reactor is shutdown (i.e., at zero power).

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

mission time: The time period that a system or component is required to operate in order to successfully perform its function.

modeling uncertainty: See *epistemic uncertainties*.

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 (for example, ''plant'' could be used to refer to a single unit or multi-unit site).

plant damage state (PDS): 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 (POS): is 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 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: The update of the PRA models to reflect plant changes such as modifications, procedure changes, or plant performance (data).

PRA upgrade: The incorporation into a PRA model of a new methodology or changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. This could include items such as new human error analysis methodology, new data update methods, new approaches to quantification or truncation, or new treatment of common cause failure.

probabilistic risk assessment (PRA): A qualitative and quantitative assessment of the risk 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 assessment, PSA).

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 [12].

radionuclide release category: See *release category*.

recovery: Restoration of a function lost as a result of a failed SSC by overcoming or compensating for its failure. Generally modeled by using HRA techniques.

release category (RC): 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 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 time at which the release begins, 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.

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

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: (1) What can go wrong? (2) How likely is it? and (3) What are the consequences if it occurs?

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

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.

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.

should: Used to state a recommendation.

shall: Used to state a mandatory requirement.

signature: A unique characteristic of the physical response of the plant to a particular accident scenario. Signatures can take many different forms, but the most common is a time-dependent plot of a calculated parameter from an integrated severe accident analysis computer code. For example, the calculated, time-dependent pressure of the reactor pressure vessel is a unique ''signature'' of the accident sequence.

significant accident progression sequence: One of the set of accident sequences contributing to large early release frequency resulting from the analysis of a specific hazard group that, when rank-ordered by decreasing frequency, sum to a specified percentage of the large early release frequency or that individually contribute more than a specified percentage of the large early release frequency for that hazard group. For this version of the standard[7], the summed percentage is 95%, and the individual percentage is 1% of the applicable hazard group (see Part 2-2.8 Requirements LE-C3, LE-C4, LE-E5, LE-C10, LE-C12, LE-D1, LE-D4, LE-D5, LE-D7, and LE-E2 of the so-called combined standard [1]). For hazard groups that are analyzed using methods and assumptions that can be demonstrated to be conservative or bounding, alternative numerical criteria may be more appropriate, and should be justified if used.

[7] Alternative criteria may be appropriate for specific applications. In particular, an alternative definition of “significant” may be appropriate for a given application where the results from PRA models for different hazard groups need to be combined.

significant accident sequence: One of the set of accident sequences resulting from the analysis of a specific hazard group defined at the functional or systemic level that, when rank-ordered by decreasing frequency, sum to a specified percentage of the core damage frequency for that hazard group or that individually contribute more than a specified percentage of core damage frequency. For this version of the standard[7], the summed percentage is 95%, and the individual percentage is 1% of the applicable hazard group (see Part 2 Requirements IE-B3, HR-H1, QU-B2, QU-C1, QU-D1, QU-D5, and QU-F2 [1]). For hazard groups that are analyzed using methods and assumptions that can be demonstrated to be conservative or bounding, alternative numerical criteria may be more appropriate, and, if used, should be justified.

significant containment challenge: A containment challenge that results in a containment failure mode that is represented in a significant accident progression sequence.

significant contributor: In the context of an accident progression sequence, a contributor that is an essential characteristic (e.g., containment failure mode or physical phenomena) of a significant accident progression sequence that would lead to the omission of the sequence if not modeled.

significant radionuclide release category: One of the set of radionuclide release categories contributing to LRF/LERF or to the overall radionuclide release frequency that, when rank-ordered by decreasing frequency, sum to 95% of the LRF/LERF or overall release frequency (excluding design basis leakage RCs) or individually contribute more than 1% of LRF/LERF or 5% of the overall release frequency.

significant release category sequence: One of the set of accident sequences contributing to a radionuclide release category frequency resulting from the analysis of a specific hazard group that, when rank-ordered by decreasing frequency, sum to a specified percentage of the release category frequency or that individually contribute more than a specified percentage of the release category frequency for that hazard group. For this version of the standard[2], the summed percentage is 95%, and the individual percentage is 1% of the applicable hazard group. For hazard groups that are analyzed using methods and assumptions that can be demonstrated to be conservative or bounding, alternative numerical criteria may be more appropriate, and should be justified if used.

source of model uncertainty: A source related to an issue in which there is no consensus approach or model and where the choice of approach or model is known to have an effect on the PRA model (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, or introduction of a new initiating event). A source of model uncertainty is labeled ‘‘key’’ when it could impact the PRA results that are being used in a decision, and consequently may influence the decision being made. Therefore, a key source of model uncertainty is identified in the context of an application. This impact would need to be significant enough that it changes the degree to which the risk acceptance criteria are met, and therefore could potentially influence the decision. For example, for an application for a licensing base change using the acceptance criteria in Regulatory Guide (RG) 1.174 [13], a source of model uncertainty or related assumption could be considered ‘‘key’’ if it results in uncertainty regarding whether the result lies in Region II or Region I or whether the result becomes close to the region boundary or not.

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 arise between sampled values when performing uncertainty analysis for cutsets consisting of basic events using a sampling approach (such as the Monte Carlo method); when taken into account, this results, for each sample, in the same value being used for all basic event probabilities to which the same data applies.

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

success path: A set of systems and associated components that can be used to bring the plant to a stable hot or cold condition and maintain this condition for at least 72 hours.

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

system failure: 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.

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.

uncertainty: A representation of the confidence in the 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 upon demand or for a prescribed time.

3. PRA CONFIGURATION CONTROL

3.1 Purpose

This section provides requirements for configuration control of a PRA to be used with this standard to support risk-informed decisions for nuclear power plants.

3.2 PRA Configuration Control Program

A PRA Configuration Control Program shall be in place. It shall contain the following key elements:

(a) a process for monitoring PRA inputs and collecting new information
(b) a process that maintains and upgrades the PRA to be consistent with the as-built, as-operated plant
(c) a process that ensures that the cumulative impact of pending changes is considered when applying the PRA
(d) a process that maintains configuration control of computer codes used to support PRA quantification
(e) documentation of the program

3.3 Monitoring PRA Inputs and Collecting New Information

The PRA Configuration Control Program shall include a process to monitor changes in the design, operation, maintenance, and industry-wide operational history that could affect the PRA. These changes shall include inputs that impact operating procedures, design configuration, initiating event frequencies, system or sub-system unavailability, and component failure rates. The program shall include monitoring of changes to the PRA technology and industry experience that could change the results of the PRA model.

3.4 PRA Maintenance and Upgrade

The PRA shall be maintained and upgraded such that its representation of the as-built, as-operated plant is sufficient to support the applications for which it is being used.

Changes in PRA inputs or the discovery of new information identified pursuant to Section 3.3 shall be evaluated to determine whether such information warrants PRA maintenance or PRA upgrade (see Section 2.2, Definitions, for the distinction between PRA maintenance and PRA upgrade). Changes that would impact risk-informed decisions shall be incorporated as soon as practical. Changes that are relevant to a specific application shall meet the SRs pertinent to that application as determined through the process described in Section 1.6 and described more fully in Section 1-3.5 of ASME/ANS RA-Sa-2009 [1].

Changes to a PRA due to PRA maintenance and PRA upgrade shall meet the requirements of the Technical Requirements section of this standard. Upgrades of a PRA shall receive a focused peer review in accordance with the requirements specified in the Peer Review section of this standard, but limited to aspects of the PRA that have been upgraded.

3.5 Pending Changes

This standard recognizes that immediately following a plant change (e.g., modifications, procedure changes, and plant performance (data)) or upon identification of a subject for model improvement (e.g., new human error analysis methodology and new data update methods), a PRA may not represent the plant until the subject plant change or model improvement is incorporated into the PRA. Therefore, the PRA configuration control process shall consider the cumulative impact of pending plant changes or model improvements on the application being performed. The impact of these plant changes or model improvements on the results of the PRA and the decision under consideration in the application shall be evaluated in a fashion similar to the approach outlined in Section 1.6 and described more fully in Section 1-3 of ASME/ANS RA-Sa-2009 [1].

3.6 Use of Computer Codes

The computer codes and associated models used to support and to perform PRA analyses shall be controlled to ensure consistent, reproducible results.

3.7 Documentation

Documentation of the Configuration Control Program and the performance of the above elements shall be adequate to demonstrate that the PRA is being maintained consistent 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 changes
(d) description of changes in a PRA due to each PRA upgrade or PRA maintenance
(e) record of the performance and results of the appropriate PRA reviews (consistent with the requirements of Section 5.6)
(f) record of the process and results used to address the cumulative impact of pending changes
(g) description of the process used to maintain software configuration control

4. TECHNICAL REQUIREMENTS

4.1 Scope

This section provides requirements for each of the technical elements that comprise 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. As noted in Section 1.3, 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 and/or external hazards addressed in ASME/ANS RA-Sa-2009 [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).**

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 seven technical elements that comprise 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 -- Accident Sequence Grouping (L1)
- *(b)* Containment Capacity Analysis (CP)
- *(c)* Severe Accident Progression Analysis (SA)
- *(d)* Probabilistic Treatment of Event Progression and Source Terms (PT)
- *(e)* Radiological Source Term Analysis (ST)
- *(f)* Evaluation and Presentation of Results (ER)
- *(g)* Interface Between Level 2 PRA and Level 3 PRA (L3)

The technical element entitled “Interface Between Level 2 PRA and Level 3 PRA (L3)” (Section 4.8) is only required if the results of the Level 2 PRA analysis will be used as input to a Level 3 consequence analysis. If the objective of the severe accident analysis ends with the determination of radionuclide releases to the environment, this technical element is not required.

4.2 Level 1/Level 2 PRA Interface–Accident Sequence Grouping

The objective of the Level 1/Level 2 PRA interface is 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.

Two essential characteristics of the interface between the Level 1 PRA and the extension to a Level 2 analysis are the following:

- *(a)* The methodology is clear, consistent with the Level 1 PRA evaluation, and creates an adequate transition from the Level 1 PRA
- *(b)* The interface boundary between the Level 1 analysis and the Level 2 analysis is defined in a manner that preserves the transfer of information (e.g., dependencies) from the Level 1 PRA to the Level 2 PRA.

One such structure to allow this interface transfer and also to provide a convenient transition point for summarizing the contributors to CDF is to consolidate or group accident sequences (or individual cut sets) 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., "Plant Damage State (PDS)" or equivalent).

4.2.1 High Level Requirements

Table 4.2-1 provides the HLRs for addressing the Level 1/Level 2 PRA interface.

Table 4.2-1 High Level Requirements for the Level 1/Level 2 PRA Interface/Accident Sequence Grouping (L1)

Designator	Requirement
HLR-L1-A	An effective interface between Level 1 analysis and Level 2 analysis shall be specified to ensure that all pertinent information required for Level 2 PRA from the Level 1 PRA is properly developed and supplemented as needed in the Level 2 PRA accident progression analysis.
HLR-L1-B	A method to transfer all necessary information (e.g., accident sequences and corresponding frequencies) from the Level 1 PRA analysis to the Level 2 PRA shall be implemented.
HLR-L1-C	Documentation of the Level 1/Level 2 PRA interface/grouping shall be consistent with the applicable supporting requirements.

4.2.2 Supporting Requirements

Tables 4.2-2, 4.2-3, and 4.2-4 provide the SRs for addressing the Level 1/Level 2 PRA interface. A set of notes that is referred to in the tables is provided at the end of Table 4.2-4.

Table 4.2-2 Supporting Requirements for HLR-L1-A

An effective interface between Level 1 analysis and Level 2 analysis shall be specified to ensure that all pertinent information required for Level 2 PRA from the Level 1 PRA is properly developed and supplemented as needed in the Level 2 PRA accident progression analysis.

<table>
<tr><th>Index No.
L1-A</th><th>Capability Category I</th><th>Capability Category II</th><th>Capability Category III</th></tr>
<tr><td>L1-A1</td><td colspan="3">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 and are necessary to effectively transfer information to the Level 2 analysis [see Note (1)]. Examples include the following:
(a) reactor coolant system (RCS) status (e.g., pressure, configuration)
(b) status of emergency core cooling systems
(c) status of containment isolation
(d) status of containment heat removal
(e) containment integrity (e.g., open, intact, vented, bypassed, or failed) [see Notes (2) and (9)]
(f) steam generator pressure and secondary water level, steam generator tube integrity (pressurized water reactors, PWRs)
(g) status of containment inerting (boiling water reactors, BWRs)
(h) containment thermodynamic conditions (such as containment pressure)
(i) availability/accessibility of mitigating equipment [see Note (7)]
(j) status of support systems (e.g., electrical power, component cooling, heating/ventilation and air conditioning (HVAC))
(k) time of core damage after the initiating event (e.g., trip or shutdown)
(l) status of other non-safety systems (e.g., control rod drive hydraulics in BWRs, service water in PWRs)
(m) environmental or physical conditions introduced by the hazard, if any, that may interfere with recovery actions that would occur after the onset of core damage
(n) initial state of fuel in the reactor [see Note (8)]
(o) design and physical configuration of primary coolant system, primary and secondary containment, and other neighboring structures, if treated
(p) physical effects of the flooding of containment and/or auxiliary building(s) on fission product release or accident mitigation (e.g., submergence of release pathway or impeding human actions)</td></tr>
</table>

Table 4.2-2 Supporting Requirements for HLR-L1-A (Cont'd)

Index No. L1-A	Capability Category I Capability Category II Capability Category III
L1-A2	IDENTIFY the accident sequence characteristics that lead to the physical characteristics identified in L1-A1. Examples include but are not limited to the following: *(a)* type of initiator and subsequent accident sequence characteristics, for example: *(i)* transients can result in high RCS pressure *(ii)* loss of coolant accidents (LOCAs) usually result in lower RCS pressure *(iii)* interfacing systems LOCAs (ISLOCAs) and steam generator tube ruptures (SGTRs) can result in containment bypass *(iv)* stuck-open steam generator secondary safety valve(s) *(v)* external hazards causing damage to accident mitigation resources such as loss of integrity of onsite storage tanks (e.g., borated water storage tank (BWST), condensate storage tank (CST), refueling water storage tank (RWST), fuel oil) *(vi)* failure of reactivity control can lead to a mismatch of energy production and heat removal *(b)* dependencies (e.g., see list in L1-B2) *(c)* status of containment safeguard systems such as sprays, fan coolers, igniters, or venting systems *(d)* status of other units on site (for multi-unit sites) and shared systems between units; the references in Note (2) provide examples of typical accident sequence characteristics
L1-A3a	IDENTIFY where the physical characteristics identified in L1-A1 and the accident sequence characteristics identified in L1-A2 are specified in the probabilistic logic model(s). For example: *(a)* which characteristics are addressed in the Level 1 PRA event trees *(b)* which characteristics are addressed in bridge trees (if applicable) *(c)* which characteristics are addressed in the CETs (or equivalent)
L1-A3b	JUSTIFY any characteristics identified in L1-A1 or L1-A2 that are excluded from the severe accident progression, containment performance, and radionuclide release categories analysis.
L1-A4	IDENTIFY plant-specific issues determined by expert judgment and/or analyses that may influence the interface between Level 1 PRA and Level 2 PRA severe accident progression analysis. Analysis support for this assessment includes deterministic calculations using computer codes or hand calculations. See SRs for HLR-SA-B for requirements on selecting appropriate computational tools [see Notes (6) and (11)].
L1-A5	Using the characteristics defined in L1-A1, L1-A2, L1-A3, and L1-A4, SPECIFY a scheme for transferring necessary input information from the Level 1 PRA accident sequences and any supplemental analyses to provide the necessary information to the Level 2 PRA.

Table 4.2-3 Supporting Requirements for HLR-L1-B

A method to transfer all necessary information (e.g., accident sequences and corresponding frequencies) from the Level 1 PRA analysis to the Level 2 PRA shall be implemented.

<table>
<tr><th>Index No. L1-B</th><th>Capability Category I</th><th>Capability Category II</th><th>Capability Category III</th></tr>
<tr><td>L1-B1</td><td colspan="3">PROVIDE a method to explicitly account for dependencies between the Level 1 PRA and Level 2 PRA models as identified in L1-A2(b). Example methods include the following:
(a) treatment in Level 2 PRA
(b) expanding Level 1 PRA [see Note (13)]
(c) construction of a bridge tree
(d) transfer of the information via PDS
(e) a combination of the above methods</td></tr>
<tr><td>L1-B2</td><td colspan="3">IDENTIFY the dependencies to be accounted for in transferring information from Level 1 PRA to Level 2 PRA logic models. Examples 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</td></tr>
<tr><td>L1-B3</td><td colspan="3">INCLUDE dependencies identified in L1-B2 in the accident progression framework.</td></tr>
<tr><td>L1-B4</td><td colspan="2">EVALUATE whether accident sequence success logic needs to be transferred to Level 2 PRA for an acceptable assessment of Level 2 PRA releases [see Note (3)].</td><td>INCLUDE Level 1 PRA accident sequence success logic in the Level 2 PRA model [see Note (3)].</td></tr>
<tr><td>L1-B5</td><td>SPECIFY sufficient accident sequence end states to provide bounding estimates of radionuclide release categories [see Note (12)].</td><td>SPECIFY sufficient accident sequence end states to capture the different contributors to significant release categories in a realistic manner such that the representative sequence in a given PDS does not vary from the other sequences in the PDS in a way that would affect the end result (e.g., the source term, the evolution of the loss of fission product barriers affecting emergency preparedness actions, or the conditional probability of releases) [see Notes (4) and (14)].</td><td>SPECIFY sufficient accident sequence end states to realistically capture the different contributors to all release categories such that the representative sequence in a given PDS does not vary from the other sequences in the PDS in a way that would affect the end result (e.g., the source term, the evolution of the loss of fission product barriers affecting emergency preparedness actions, or the conditional probability of releases) [see Notes (4) and (5)].</td></tr>
</table>

Table 4.2-3 Supporting Requirements for HLR-L1-B (Cont'd)

Index No. L1-B	Capability Category I	Capability Category II	Capability Category III
L1-B6	Conservatively GROUP accident sequences to transfer information (defined in L1-A) to the Level 2 PRA [see Note (12)]. 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.	GROUP accident sequences into a sufficient set to realistically model the PDS dependencies and other plant conditions that are needed to represent the significant accident progression sequences in the Level 2 PRA logic model.	ESTABLISH the transfer of information (dependencies, plant conditions) from Level 1 PRA to Level 2 PRA to allow the realistic representation of all accident progression sequences in the Level 2 PRA logic model.
L1-B7	ENSURE that the grouping process into PDS (or other interface issues) does not result in screening out (i.e., prematurely truncating) accident sequences that are important in the characterization of the radionuclide release (e.g., significant radionuclide release categories) or sequences that defeat all or most containment mitigation measures [see ER-C1].	TRANSFER the total CDF from the Level 1 PRA to the Level 2 PRA [see Note (10)].	TRANSFER the total CDF from the Level 1 PRA to the Level 2 PRA including the uncertainty distributions on the Level 1 PRA cut sets/sequences [see Note (10)].

Table 4.2-4 Supporting Requirements for HLR-L1-C

Documentation of the Level 1/Level 2 PRA interface/grouping shall be consistent with the applicable supporting requirements.

Index No. L1-C	Capability Category I	Capability Category II	Capability Category III
L1-C1	DOCUMENT the Level 1/Level 2 PRA interface/grouping of sequences in a manner that facilitates PRA applications, upgrades, and peer review.		
L1-C2	DOCUMENT Level 1 PRA attributes that are considered in the Level 1 PRA/Level 2 PRA interface.		
L1-C3	DOCUMENT the assumptions and methods used to propagate information across the Level 1 PRA/Level 2 PRA interface and address dependencies.		

NOTES:

(1) 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 the efficiency of fission product deposition within the RCS. The status of reactor coolant injection can be relevant for evaluating opportunities for core cooling, but can also affect RWST inventory for containment cooling and the presence of water in the vessel cavity at the time of lower head failure. 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, 2010 [14].

(2) References:

(a) Nuclear Power Plant Response to Severe Accidents, IDCOR Technical Summary Report, Technology for Energy Corp. 1984 [15]

(b) NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," December 1990 [16]

(c) NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," December 1997 [17]

(d) EPRI Report 1022186, January 2010 [18]

(3) 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 cut sets 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 of the 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 <u>same</u> question is asked in the Level 2 PRA.

(4) Examples of the factors that will affect the PDS variation are those given in SRs L1-A1 and L1-A2 as well as the considerations listed in L1-A4.

(5) In this context, the term integrated model refers to the ability of the Level 2 PRA model to directly extract the underlying information (e.g., basic events) from the Level 1 PRA model. Such a model may or may not use sequence grouping. As an example to support L1-B5 Capability Category III, a single integrated Level 1/2 PRA model may be used. The transfer of information from Level 1 PRA to Level 2 PRA is carried out by an integrated Level 1/Level 2 PRA model that accurately transfers dependencies within the model. For this approach, the grouping of accident sequences may be performed for any of the following example reasons: (a) as a display method that allows examining some or all functional accident group (or PDS) results at the intermediate point of core damage; (b)

to maintain correct house event settings; (c) to transfer to the correct CET logic structure (changes to CETs are not necessarily implemented only as logic switches as there may be hardwired differences); and (d) to preserve initiating event characteristics that impact the logic models.

(6) 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 resulting 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 require 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 reactor pressure vessel (RPV) when SRVs are operating to depressurize the RPV (e.g., valves with pneumatic, electro-magnetic, etc. operators).
  - *(d)* Containment spray function including the use of portable pumps.

  The identification of plant specific interface issues may arise during the development of all aspects of the Level 2 PRA (e.g., CET development).

(7) Consider that initiating events and Level 1 PRA accident sequence successes and failures may influence the availability of and accessibility to mitigating equipment.

(8) 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.

(9) 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.

(10) The total CDF is transferred from the Level 1 PRA to the Level 2 PRA in CC-II and CC-III. 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.

(11) Guidance on the use of expert judgment (if applied) is available in Section 1.7.3.

(12) As an example to support 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 requires conservative modeling of the PDS dependencies and other plant conditions on the Level 2 accident progression modeling to adequately cover the dependencies 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 release categories toward those representing earlier and/or larger releases of fission products.

(13) 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.

(14) As an example to support L1-B5 Capability Category II, the number of PDSs selected are sufficient to realistically transfer dependencies and plant conditions for the representation of significant accident progression sequences in the logic model and supporting deterministic models.

4.3 Containment Capacity Analysis

The objective of this section is to define the containment capacity to withstand severe accident progression challenges.

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.

4.3.1 High Level Requirements

HLRs for primary containment capacity analysis are listed in Table 4.3.1.

Table 4.3-1 High Level Requirements for Containment Capacity Analysis (CP)

Designator	Requirement
HLR-CP-A	The mechanisms of primary containment failure shall be identified as input to the assessment of containment capacity.
HLR-CP-B	A method (or methods) shall be selected to evaluate structural capacity to withstand postulated loads and challenges.
HLR-CP-C	The capacity of the primary containment pressure boundary to withstand loads generated by external hazards, containment challenges evolving prior to core damage (e.g., loss of containment heat removal), and containment challenges generated by core damage accidents shall be determined.
HLR-CP-D	Uncertainties in primary containment failure analysis shall be identified.
HLR-CP-E	Documentation of the assumptions, models used, and results of the primary containment capacity analysis shall be consistent with the applicable supporting requirements.

4.3.2 Supporting Requirements

Tables 4.3-2 through 4.3-6 provide SRs for containment capacity analysis. A set of notes that is referred to in these tables is provided at the end of Table 4.3-6.

It is expected that the requirements delineated in these tables 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 “failure mechanism” is clearly understood in the context of structural failure 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 supporting requirements listed in this subsection (Tables 4.3-2 through 4.3-6) should be applied for each of these conditions, as delineated in the Level 1 PRA results for a particular POS.

Table 4.3-2 Supporting Requirements for HLR-CP-A

The mechanisms of primary containment failure shall be identified as input to the assessment of containment capacity.

Index No. CP-A	Capability Category I	Capability Category II	Capability Category III
CP-A1	PERFORM a plant-specific search to identify plausible failure mechanisms, accounting for unique design features and guided by calculating environmental conditions within containment during representative severe accident sequences. In developing a list of potential containment failure mechanisms, INCLUDE relevant failure mechanisms from a standardized list of “typical” failure mechanisms from studies of other plants with similar containment design features. For example, include the relevant failure mechanisms from CP-A3 through CP-A8.		
CP-A2	JUSTIFY screening out failure mechanisms that are developed in CP-A1.		
CP-A3	INCLUDE failure mechanisms of the global containment membrane such as closure doors, hatches, mechanical penetrations, electrical assemblies, and bellows seals [see Note (4)].		
CP-A4	If core damage accident sequences initiated by external hazards are included in the Level 2 PRA, INCLUDE containment failure mechanisms caused by the evaluated external hazards.		
CP-A5	INCLUDE containment failure mechanisms caused by severe accident phenomena. Examples of these phenomena include hydrogen combustion (deflagration and detonation), material creep, or seal failure due to sustained exposure to high temperatures, structural consequences of hydrodynamic loads, dynamic interactions between molten core debris and water, direct contact between core debris and containment structures, concrete cracking, liner tearing, and radiation damage to containment sealant materials [see Notes (6) and (9)].		
CP-A6	INCLUDE indirect mechanisms of containment failure caused by severe accident phenomena, for example: *(a)* erosion or displacement of structures internal to the containment causing a loss of containment integrity *(b)* a seismic event *(c)* failure of the reactor vessel lower head at high pressure *(d)* thermo-chemical erosion of a concrete reactor pedestal that might result in displacement of the reactor pressure vessel *(e)* movement of appended piping and structural damage to piping penetrations in the containment pressure boundary		
CP-A7	For containment designs including pressure-suppression water pools, INCLUDE hydrodynamic challenges to containment integrity caused by a high-pressure blowdown of steam and/or non-condensable gases from the RCS into the suppression pool (or equivalent) [see Notes (1) and (10)].		

Table 4.3-2 Supporting Requirements for HLR-CP-A (Cont'd)

Index No. CP-A	Capability Category I	Capability Category II	Capability Category III
CP-A8	IDENTIFY a generic quantitative estimate of the likelihood of pre-existing failure modes or plant conditions that compromise containment capability to withstand severe accident challenges. Data can include industry experience regarding results of 10CFR50 Appendix J testing on containment and penetrations and 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 (refer to CP-B4) [see Note (13)].		IDENTIFY a plant-specific quantitative assessment of the likelihood of pre-existing failure modes or plant conditions that compromise containment capability to withstand severe accident challenges. Calculations may use plant-specific data to Bayesian update the generic pre-existing failure modes. Examples include: plant experience with results of 10CFR50 Appendix J testing on containment and penetrations and 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 (refer to CP-B4) [see Note (13)].
CP-A9	If buildings outside the containment pressure boundary (e.g., Reactor Building or Auxiliary Building) are assumed to participate in the release pathway for fission products released to the environment, IDENTIFY potential failure mechanisms that could compromise the structural integrity of these buildings as a consequence of severe accident progression (e.g., hydrogen release and combustion).		
CP-A10	IDENTIFY those failure mechanisms assessed in CP-A1 through CP-A9 that are to be addressed in the assessment of containment capacity in HLR CP-B.		

Table 4.3-3 Supporting Requirements for HLR-CP-B

A method (or methods) shall be selected to evaluate structural capacity to withstand postulated loads and challenges.

Index No. CP-B	Capability Category I	Capability Category II	Capability Category III
CP-B1	ESTIMATE the ultimate strength of the containment pressure boundary for both rupture and excess leakage failure modes identified in CP-A with a method that uses existing design basis containment pressure calculations or similar information to estimate the containment failure pressure. For example, apply a scalar to the containment design pressure based on observed ratios of failure-to-design pressure calculations for other similar containment structures. Justify any ratio greater than 2.0 [see Note (3)]. OR USE a computational method that implements a generic model and is justified for use at the specific plant. For example, quantitative results of reference plant calculations can be adapted to the plant and accident scenario of interest. JUSTIFY the use of reference plant analyses to the plant and accident conditions to which they are applied.	CALCULATE the ultimate strength of the containment pressure boundary for both rupture and excess leakage failure modes identified in CP-A using a method that relies 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 conditions. OR USE results of applicable experimental measurements of containment performance for specific failure mechanisms. JUSTIFY the use of experimental measurements to the plant and accident conditions to which they are applied. OR USE a combination of the above methods.	CALCULATE the ultimate strength of the containment pressure boundary for both rupture and excess leakage failure modes identified in CP-A using a method that relies on a validated [see Note (12)], plant-specific, three-dimensional, finite-element, non-linear structural model that explicitly incorporates major geometric discontinuities and constraints for the assessment of ultimate pressure capacity such as large hatches, penetrations, and anchors.

Table 4.3-3 Supporting Requirements for HLR-CP-B (Cont'd)

Index No. CP-B	Capability Category I	Capability Category II	Capability Category III
CP-B2	DETERMINE capacity limit(s) for structural materials assuming constant temperature material properties as part of the method selected to evaluate the structural capacity of the containment or other buildings. For example, "failure" might be defined as a maximum global membrane strain away from discontinuities of one percent for the assessment of ultimate pressure capacity for cylindrical reinforced concrete containments. Note that multiple capacity limits might be defined depending on the number and type of failure mechanisms considered. For example, cracking of concrete containments versus catastrophic rupture and tearing of a steel liner versus leakage through penetration seals.		DETERMINE and JUSTIFY capacity limit(s) for structural materials using temperature-dependent material properties. Note that multiple capacity limits might be defined depending on the number and type of failure mechanisms considered. For example, cracking of concrete containments versus catastrophic rupture and tearing of a steel liner versus leakage through penetration seals.
CP-B3	DETERMINE the "as built" or "as designed" structural geometry and material composition to be used as the basis of the initial assessment to be performed in CP-B4.		
CP-B4	CHARACTERIZE the impact of material deterioration for structures with more than 10 years of service [see Note (2)].	EVALUATE the impact of material deterioration for structures with more than 10 years of service [see Note (2)].	
CP-B5	SPECIFY a bounding quasi-static thermal-mechanical load or the physical attributes of challenges on the containment structure used to evaluate containment capacity [see Note (6)].	SPECIFY plant-specific realistic quasi-static thermal-mechanical loads or the physical attributes of challenges on the containment structure used to evaluate containment capacity for significant accident progression sequences.	

Table 4.3-3 Supporting Requirements for HLR-CP-B (Cont'd)

Index No. CP-B	Capability Category I	Capability Category II	Capability Category III
CP-B6	ASSUME buildings outside containment structures (e.g., BWR Reactor Building, PWR Auxiliary Building, Turbine Buildings) do not have the capacity to withstand thermal or mechanical loads generated during core damage accident sequences.	DETERMINE the capacity of 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	USE a representative load or challenge with one independent variable. For example, in the assessment of ultimate pressure capacity, internal temperature might be fixed as the design basis maximum; however, pressure is larger than design basis by an unknown amount.	USE a series of discrete static conditions of multiple variables affecting capacity. For example, in the assessment of ultimate pressure capacity, discrete combinations of pressure and temperature are defined for analysis based on plant-specific calculations of severe accident progression for representative sequences.	USE a series of discrete static conditions of multiple variables affecting capacity. For example, in the assessment of ultimate pressure capacity, discrete combinations of pressure and temperature are defined for analysis based on plant-specific calculations of severe accident progression for representative sequences. USE discrete static multivariate boundary conditions with estimates of exposure time at each value.

Table 4.3-4 Supporting Requirements for HLR-CP-C

The capacity of the primary containment pressure boundary to withstand loads generated by external hazards, containment challenges evolving prior to core damage (e.g., loss of containment heat removal), and containment challenges generated by core damage accidents shall be determined.

Index No. CP-C	Capability Category I	Capability Category II	Capability Category III
CP-C1 [Note (7)]	For each failure mechanism and operational mode under consideration, SPECIFY bounding (minimum) thresholds for failure.	For each failure mechanism and operational mode under consideration, SPECIFY realistic thresholds for failure as a function of discrete combinations of independent variables (e.g., temperature and pressure).	

Table 4.3-4 Supporting Requirements for HLR-CP-C (Cont'd)

Index No. CP-C	Capability Category I	Capability Category II	Capability Category III
CP-C2 [Note (7)]	For each failure mechanism and operational mode under consideration, conservatively SPECIFY the location and representative value of the final opening size in the containment pressure boundary if its failure criterion is met [see Note (11)].	For each failure mechanism and operational mode under consideration, SPECIFY the location and a realistic value of the final opening size in the containment pressure boundary as a function of pressure if its failure criterion is met for significant containment challenges. If multiple (alternate) failure locations and/or opening sizes are considered, SPECIFY conditional probabilities assigned to each possibility for significant containment challenges.	For each failure mechanism and operational mode under consideration, SPECIFY the location and a realistic description of the opening size as a function of the pressure and temperature load on the containment boundary. For example, leak rate or area in a concrete structure might begin as cracks at an elevated pressure and grow to a larger area if pressure continues to increase. Similarly, the opening in a steel containment due to direct contact with molten core debris might begin as a small opening at the initial point of contact and increase to a larger size with sustained exposure. If multiple (alternate) failure locations and/or opening sizes are considered, SPECIFY conditional probabilities assigned to each possibility for all quantified containment challenges.
CP-C3	If external hazards are included in the Level 2 PRA, ESTIMATE the capacity of the containment to withstand the external hazard. For example, bounding estimates may be used to relate containment failure modes and site peak ground accelerations [see Note (5)].	If external hazards are included in the Level 2 PRA, CALCULATE the response of the containment pressure boundary to each such hazard (i.e., CALCULATE hazard-specific fragilities) including interactions with major penetrations, in conformance with the applicable requirements delineated in ASME/ANS RA-Sa-2009 [1] relating to fragilities (e.g., Section 5-2.2).	

Table 4.3-5 Supporting Requirements for HLR-CP-D

Uncertainties in primary containment failure analysis shall be identified.

Index No. CP-D	Capability Category I	Capability Category II	Capability Category III
CP-D1	IDENTIFY sources of parameter uncertainty, modeling uncertainty, and assumptions used in the deterministic analysis of containment failure [see Note (8) for examples].		
CP-D2	CHARACTERIZE the uncertainty range in thresholds for containment failure using engineering judgment [see Note (7)].	CHARACTERIZE the uncertainty in containment failure criteria in the form of a probability density function (fragility curve) [see Note (7)].	
CP-D3	CHARACTERIZE the uncertainty range in the final opening size of containment failure using engineering judgment.	CHARACTERIZE the uncertainty range in the final opening size of containment to permit a characterization of uncertainties in applications using structured sensitivity analysis.	
CP-D4	For each source of model uncertainty and related assumption identified in CP-D1, CHARACTERIZE how the containment strength or resistance to failure is affected [see Note (8) for examples].		

Table 4.3-6 Supporting Requirements for HLR-CP-E

Documentation of the assumptions, models used, and results of the primary containment capacity analysis shall be consistent with the applicable supporting requirements.

Index No. CP-E	Capability Category I	Capability Category II	Capability Category III
CP-E1	DOCUMENT the containment failure capacity analysis in a manner that facilitates PRA applications, upgrades, and peer review.		
CP-E2	DOCUMENT the mechanisms of containment failure. This documentation typically includes the following: *(a)* geometric configuration(s) *(b)* material composition *(c)* type and extent of material or geometric degradation due to adverse environmental conditions (e.g., corrosion, concrete decomposition, etc.) *(d)* metrics to define what was screened out from consideration in the PRA		
CP-E3	DOCUMENT the failure criteria (thresholds or fragility curve) defined for each mechanism and support each criterion with a technical justification.		
CP-E4	DOCUMENT the failure criteria (thresholds or fragility curve) and technical rationale defined for each failure mechanism.		
CP-E5	DOCUMENT the technical basis for the location and opening size (or leak rate) resulting from each failure mechanism and the technical basis for the probabilities used to characterize uncertainty.		
CP-E6	DOCUMENT the characterization of the sources of model uncertainty and related assumptions (as identified in CP-D1 through CP-D4).		

NOTES:

(1) Non-condensable gases may include air, containment atmosphere inerting gas, and hydrogen and other non-condensable gases generated by in-vessel oxidation of metallic core components or ex-vessel (e.g., core concrete interactions).

(2) A discussion of methodology for including degraded conditions in a containment structural analysis and example analyses of various degraded (aged) containment designs can be found in NUREG/CR-6920 [19].

(3) The need to justify extrapolation greater than a factor of two is based on the concern that as the structure transitions farther above its design pressure, non-linear factors and structural discontinuities (e.g., large penetrations) may become important factors in defining the ultimate containment strength. Thus, extrapolations beyond this range should require additional justification.

(4) Failure of containment isolation due to the inability of valves to close is not considered in the scope of technical element CP.

(5) For seismic events in particular, consider that relative motions between the Auxiliary Building (or equivalent) and containment can overstress containment penetrations and cause localized failures.

(6) Thermal-mechanical challenges are based on the results of severe accident analyses (see Section 4.4).

(7) Note that for some modes of reactor operation, failure limits may differ from those at-power 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.

(8) 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, etc.
(9) Severe accident conditions may impose high temperature and radiation challenges on seals and sealants.
(10) Hydrodynamic 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 RWSTs that may also lead to the consideration of challenges related to hydrodynamic loads.
(11) 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.
(12) Applicable experimental data should be considered in the validation of the plant-specific model.
(13) The manner in which these effects are considered in the probabilistic model is the subject of SR PT-C8.

4.4 Severe Accident Progression Analysis

The objective of Severe Accident Progression Analysis to support a Level 2 PRA is 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. 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 areas in which the Severe Accident Progression Analysis supports Level 2 PRA include the following:

- describing the chronology (timeline) of postulated accidents involving significant damage to reactor fuel;
- characterizing thermal, chemical, and mechanical challenges to engineered barriers to fission product release to the environment; and
- generating estimates of radionuclide release to the environment for accident sequences identified as contributors to the frequency of release.

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 Section 4.5.

4.4.1 High Level Requirements

Table 4.4-1 provides the HLRs for severe accident progression analysis.

Table 4.4-1 High Level Requirements for Severe Accident Progression Analysis (SA)

Designator	Requirement
HLR-SA-A	The objectives of the calculations shall be defined, and the quantitative parameters or metrics of severe accident behavior that deterministic analysis will calculate shall be identified.
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 (computational) model shall be selected for generating each estimate of the quantitative parameters defined in HLR-SA-A.
HLR-SA-D	Calculations shall be performed as needed to support the probabilistic accident progression framework (see Section 4.5).
HLR-SA-E	The effects of uncertainties in calculating plant response to severe accidents shall be characterized.
HLR-SA-F	Documentation of the Severe Accident Progression Analysis shall be consistent with the applicable requirements

4.4.2 Supporting Requirements

Tables 4.4-2 through 4.4-7 provide the SRs for severe accident progression analysis. A set of notes that are referred to in the tables is provided at the end of Table 4.4-7.

Table 4.4-2 Supporting Requirements for HLR-SA-A

The objectives of the calculations shall be defined, and the quantitative parameters or metrics of severe accident behavior that deterministic analysis will calculate shall be identified.

Index No. SA-A	Capability Category I	Capability Category II	Capability Category III
SA-A1	DEFINE the objectives of deterministic analysis performed to support the Level 2 PRA. Examples include but are not limited to: *(a)* determine order/timing of severe accident events *(b)* determine order/timing of operator actions *(c)* determine Level 2 PRA success criteria *(d)* generate quantitative measures of parameters used to estimate probabilities of uncertain events and phenomena *(e)* calculate the plant responses that could challenge the ultimate capacity of the containment or affect equipment survivability or accessibility *(f)* calculate source terms (e.g., radionuclide release category characteristics)		
SA-A2	SPECIFY the output parameters to be calculated; examples include: *(a)* RCS pressure and temperature *(b)* containment pressure and temperature [see Note (3)] *(c)* criticality *(d)* water levels (RPV, containment, CST, BWST, RWST)		

Table 4.4-3 Supporting Requirements for HLR-SA-B

Assumptions used to perform deterministic calculations shall be identified, and values of input parameters shall be estimated.

Index No. SA-B	Capability Category I	Capability Category II	Capability Category III
SA-B1	LIST assumptions and sources of uncertainty used in performing deterministic calculations.		
SA-B2	ESTIMATE values of input parameters for deterministic calculations using conservative methods and assumptions [see Note (5)]. Examples of input parameters include: *(a)* containment failure pressure and temperature *(b)* induced RCS failure criteria *(c)* reactor vessel failure criteria *(d)* fission product inventories in the core at the time of core damage	ESTIMATE realistic values of input parameters for deterministic calculations. Examples of input parameters include: *(a)* containment failure pressure and temperature *(b)* induced RCS failure criteria *(c)* reactor vessel failure criteria *(d)* fission product inventories in the core at the time of core damage	

Table 4.4-4 Supporting Requirements for HLR-SA-C

An appropriate deterministic (computational) model shall be selected for generating each estimate of the quantitative parameters defined in HLR-SA-A.

Index No. SA-C1	Capability Category I	Capability Category II	Capability Category III
SA-C1	USE a reference plant calculation or calculation derived from first principles and/or well-established correlations that provides a conservative estimate for the calculated results [see Notes (5) and (7)].	USE a realistic modeling tool that reflects plant-specific design features and contemporary knowledge of severe accident behavior (that is, validated against available experimental data or other established benchmarks) [see Note (6)].	
SA-C2	SPECIFY a basis for applying the method selected in 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. A qualitative evaluation of a relevant application of the selected method that has been used for a similar class of plant (e.g., Owner's Group generic study) may be used.		

Table 4.4-4 Supporting Requirements for HLR-SA-C (Cont'd)

Index No. SA-C1	Capability Category I	Capability Category II	Capability Category III
SA-C3	JUSTIFY method(s) used to adapt or modify results of a reference calculation.	JUSTIFY selections of modeling options (e.g., alternative correlations or models within a computer code) and values of input parameters applied to the modeling tool selected in SA-C1.	

Table 4.4-5 Supporting Requirements for HLR-SA-D

Calculations shall be performed as needed to support the probabilistic accident progression framework (see Section 4.5).

Index No. SA-D	Capability Category I	Capability Category II	Capability Category III
SA-D1	ADAPT quantitative results of reference plant calculations to estimate the parameters defined in SA-A2.	CALCULATE parameters defined in SA-A2 for significant accident progression sequences.	CALCULATE parameters defined in SA-A2 for all accident progression sequences.
SA-D2	DEMONSTRATE similarity in reactor and/or containment design between the reference plant and the plant being analyzed, as appropriate, by comparing plant-design or operating characteristics that influence the calculated condition, process, or event of interest. For example, 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.	DEMONSTRATE reasonableness and acceptability of the calculated results defined in SA-A2. Examples of methods to achieve this include: *(a)* comparison with results of the same analyses performed for similar plants, accounting for differences in unique plant features *(b)* comparison with results of similar analyses performed with other plant-specific codes *(c)* check by other means appropriate to the particular analysis	
SA-D3	SPECIFY and JUSTIFY the end-point or termination time of severe accident calculations. For the purpose of source term evaluation, USE a minimum end-point or termination time of 36 hours after the onset of core damage (and containment has reached a stable configuration) for all severe accident calculations [see Note (4)].		

Table 4.4-6 Supporting Requirements for HLR-SA-E

The effects of uncertainties in calculating plant response to severe accidents shall be characterized.

Index No. SA-E	Capability Category I	Capability Category II	Capability Category III
SA-E1	DEVELOP a list of modeling uncertainties and assumptions from published contemporary resources describing known severe accident modeling uncertainties and assumptions for similar reactor/containment designs [see Note (1)].	IDENTIFY uncertain models or assumptions that affect severe accident progression and/or radionuclide source terms, for significant accident progression sequences.	IDENTIFY uncertain models or assumptions that affect severe accident progression and/or radionuclide source terms.
SA-E2	IDENTIFY input parameters particular to the modeling tool selected in SA-C1 that reflect the uncertain models or assumptions defined in SA-E1.	DEFINE variations in input parameters particular to the modeling tool selected in SA-C1 that reflect the uncertain models or assumptions defined in SA-E1 [see Note (2)].	
SA-E3	For significant accident progression sequences, CHARACTERIZE the effects of uncertainties associated with input parameters.	For significant accident progression sequences with uncertain models or assumptions, PERFORM sensitivity analyses to evaluate the effects of uncertainties associated with calculation input parameters.	PERFORM sensitivity analyses to evaluate the effects of uncertainties associated with calculation input parameters.
SA-E4	IDENTIFY sources of model uncertainties and assumptions identified in SA-E1 that are not investigated in SA-E3.		
SA-E5	For each source of model uncertainty and related assumption identified in SA-E4, CHARACTERIZE how the accident progression analysis results are affected.		

Table 4.4-7 Supporting Requirements for HLR-SA-F

Documentation of the Severe Accident Progression Analysis shall be consistent with the applicable supporting requirements.

Index No. SA-F	Capability Category I	Capability Category II	Capability Category III
SA-F1	DOCUMENT deterministic severe accident analysis in a manner that facilitates PRA applications, upgrades, and peer review.		

Table 4.4-7 Supporting Requirements for HLR-SA-F (Cont'd)

Index No. SA-F	Capability Category I	Capability Category II	Capability Category III
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 shall be provided for an independent person to reproduce the input data from original sources, e.g., description of basis for input selection and basis for analyses and discussion of code limitations and assumptions.	
SA-F3	DOCUMENT alternative modeling assumptions and/or values of uncertain input parameters used in sensitivity and/or uncertainty analysis.		
SA-F4	DOCUMENT the characterization of the sources of model uncertainty and related assumptions (as identified in SA-E3 and SA-E5).		
SA-F5	DOCUMENT results of calculations, including: *(a)* tabular summaries of the order and timing of important events *(b)* graphical displays (e.g., plots, graphs) showing temporal signatures of important calculated parameters		
SA-F6	DOCUMENT evaluations of the reasonableness of calculated results. Example contents of documentation include: *(a)* comparisons to references, standards, or sensitivity calculations *(b)* independent calculations used to confirm results *(c)* explanations of observed trends and counterintuitive results		

NOTES:

(1) Example documents describing known uncertainties in severe accident progression include: NUREG-1855 [20] and companion EPRI documents [21, 22], contemporary NRC Research documents on severe accident modeling [23, 24], IAEA Level 2 PSA guidance [14, 25], and European Consensus documents on Severe Accident Management and Level 2 PSA [26, 27].

(2) Input parameters might not be available to investigate the effects of some modeling uncertainties and/or assumptions identified in SA-E1.

(3) Includes drywell and suppression pool for BWRs.

(4) Justification of end-point/termination time would typically address trends in results at the termination time and provide a technical basis for claims that results and conclusions drawn from the calculation would not change if the termination time was extended.

(5) Conservative in this context implies that the selected values for input parameters (or choice of models) can be demonstrated to result in an earlier, larger, or both earlier and larger release of fission products to the environment than would result from a more realistic choice of parameters or models.

(6) Reference [18] is a useful resource for investigating the validation of computational models to experimental data.

(7) For these assessments, a reference plant calculation is a calculation performed for a reference plant that is sufficiently close to the specific plant being evaluated so that the results are found to be approximately correct.

A first principles calculation is one that is developed from basic physics and chemistry equations that approximates the complex interactions to be modeled.

4.5 Probabilistic Treatment of Accident Progression and Source Terms

The objective of the probabilistic treatment of Level 2 PRA accident progression and source terms is 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 methodology is clear and consistently linked with the Level 1 PRA evaluation to create an adequate transition from the Level 1 PRA;
(b) Human actions, mitigation systems, and phenomenological behaviors that can alter the event progression are adequately evaluated and characterized;
(c) Dependencies are appropriately reflected in the model structure;
(d) Phenomenology is appropriately characterized and modeled;
(e) Analyses are provided to support equipment success criteria, time windows for human action, access requirements for human actions, and other recoveries;
(f) Level 2 PRA end states are defined in sufficient detail so that they can be characterized in terms of radionuclide release timing, containment failure mode, radionuclide release distribution and magnitude; and
(g) The frequency of the severe accident sequences leading to the defined end states is calculated.

4.5.1 High Level Requirements

Table 4.5-1 provides the HLRs for the probabilistic treatment of accident progression and source terms.

Table 4.5-1 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 release (source term) categories that, in turn, are capable of distinguishing sequences with significantly different radiological consequences.
HLR-PT-B	Branching probabilities (split fractions) or supporting models for quantitatively characterizing severe accident phenomena shall be developed.
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 developed.
HLR-PT-D	Branching probabilities (split fractions) or supporting models for characterizing the reliability of human actions in the accident progression framework shall be developed.
HLR-PT-E	The frequencies of radionuclide release categories using appropriate models and codes, and accounting for method-specific limitations and features shall be calculated.
HLR-PT-F	The probabilistic treatment of event progression and source terms consistent with the applicable supporting requirements shall be documented.

4.5.2 Supporting Requirements

Tables 4.5-2 through 4.5-7 provide the SRs for the probabilistic treatment of event progression and source terms. A set of notes that are referred to in the tables is provided at the end of Table 4.5-7.

Table 4.5-2 Supporting Requirements for HLR-PT-A

An accident progression framework shall be developed that supports the grouping of severe accident sequences into radionuclide release (source term) categories that, in turn, are capable of distinguishing sequences with significantly different radiological consequences.

Index No. PT-A	Capability Category I	Capability Category II	Capability Category III
PT-A1	USE a methodology for representing the severe accident progression associated with each Level 1 PRA accident sequence or PDS and for quantifying the frequency of different potential accident progressions. This may include a single integrated Level 1PRA/Level 2 PRA as chosen in L1-B4.		
PT-A2	INCLUDE the following attributes in the CET (or equivalent): *(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)* explicit 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 conservation of accident sequence frequency from the initiating event to the summed frequency of all possible end-states *(d)* aggregation of individual accident progressions into groups (release categories) 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)* recoveries including those that could have negative impacts (for example, de-inerting due to the recovery of sprays)		
PT-A3	DEVELOP a logic structure using the method selected in PT-A1 (i.e., CET or equivalent) that incorporates the items identified in PT-A2, PT-A4, PT-A6, and PT-A7 as well as: *(a)* initial conditions of Level 2 analysis (output from Level 1 PRA/Level 2 PRA interface, Section 4.3.2) *(b)* discrimination of different radiological release pathways to the environment *(c)* restoration of coolant injection function prior to RPV lower head failure, offering the possibility of terminating core damage within the RPV *(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 4.5-8) *(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-A4	SPECIFY 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 4.5-8. Additional resources are provided in Note (8).		

Table 4.5-2 Supporting Requirements for HLR-PT-A (Cont'd)

Index No. PT-A	Capability Category I	Capability Category II	Capability Category III
PT-A5	JUSTIFY the exclusion of any phenomenological events identified in PT-A4 from the model.		
PT-A6	ASSUME that mitigating actions by plant personnel such as those described in SAMGs are not taken.	INCLUDE events in the CET (or equivalent) that represent the effects of mitigating actions by plant personnel directed by plant-specific SAMGs or proceduralized actions, accounting for their effect on radionuclide release for significant accident progression sequences.	INCLUDE events in the CET (or equivalent) that represent the effects of mitigating actions by plant personnel directed by plant-specific SAMGs or proceduralized actions (including errors of commission known to have a significant adverse impact), accounting for their effect on radionuclide release for accident progression sequences with frequency above the truncation limit.
PT-A7	ASSUME that potential mitigating effects of structures outside the containment pressure boundary are not effective for source term attenuation.	INCLUDE events in the CET (or equivalent) that reflect accident behavior within structures outside the containment pressure boundary and/or the response of mitigating systems outside the containment pressure boundary and affect source term attenuation for significant accident progression sequences.	INCLUDE events in the CET (or equivalent) that reflect accident behavior within structures outside the containment boundary and/or the response of mitigating systems outside the containment pressure boundary and affect source term attenuation for accident progression sequences with frequency above the truncation limit.
PT-A8	ASSUME that benevolent failures of active components do not occur.	IDENTIFY any credit taken for benevolent failures of active components in significant accident progression sequences.	
PT-A9	INCLUDE capability to determine importance measures of individual PDSs to RCs.	INCLUDE capability to determine importance measures of individual systems from the Level 1 PRA analysis to RCs.	INCLUDE capability to determine importance measures of individual components from the Level 1 PRA analysis to RCs.

Table 4.5-2 Supporting Requirements for HLR-PT-A (Cont'd)

Index No. PT-A	Capability Category I	Capability Category II	Capability Category III
PT-A10	In the CET (or equivalent), MODEL the logical dependencies between systems, components or human actions in the Level 1 PRA and event headings in the CET 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. Example areas of logical dependency include but are not limited to the following: *(a)* development of adverse environment *(b)* human actions *(c)* survivability of structures, systems, and components (SSCs) *(d)* alternating current (AC) power restoration	INCLUDE in the CET (or equivalent) logic that accounts for dependencies arising from the Level 1 PRA or between different headings in the CET realistically for significant accident progression sequences [see examples under Capability Category I].	INCLUDE in the CET (or equivalent) logic that accounts for dependencies arising from the Level 1 PRA or between different headings in the CET realistically using a fully coupled Level 1/Level 2 PRA model [see examples under Capability Category I].
PT-A11	DEVELOP model logic that generates a conservative assessment of the frequency of accident progression sequences resulting in large and early radionuclide release [see Notes (7) and (9)].	DEVELOP model logic necessary to provide a realistic assessment of accident progression frequency for significant accident progression sequences [see Note (7)].	DEVELOP model logic necessary to provide a realistic assessment of accident progression frequency for accident progression sequences with frequencies above the truncation limit [see Note (7)].
PT-A12	INCLUDE events in the CET (or equivalent) that represent operator actions required to establish containment closure during accident sequences with an open containment pressure boundary (e.g., during shutdown conditions), taking into account the time available and time required for closure.		
PT-A13	SPECIFY end states using the definitions and attributes of RCs described in Section 4.6.		
PT-A14	IDENTIFY intact end states and CHARACTERIZE the leakage RCs.		
PT-A15	IDENTIFY the RC for each of the Level 2 PRA accident sequences (or equivalent) [refer to HLR ST-A].		
PT-A16	JUSTIFY any generic or plant-specific calculations or references used to categorize releases as non-LERF/non-large release (LRF) contributors based on release magnitude or timing [refer to SR ST-E3].		

Table 4.5-3 Supporting Requirements for HLR-PT-B

Branching probabilities (split fractions) or supporting models for quantitatively characterizing severe accident phenomena shall be developed.

Index No. PT-B	Capability Category I	Capability Category II	Capability Category III
PT-B1	SELECT a method (e.g., expert judgment, parametric analysis) for defining numerical values of probability to reflect epistemic (modeling) uncertainty in phenomenological events [see Note (11)].		
PT-B2	JUSTIFY the rationale for using different methods to calculate or assign numerical values of probability if more than one method is used in the PRA.		
PT-B3	USE conservative boundary conditions to estimate branching probabilities (split fractions) for phenomenological events [see Note (9)].	USE realistic boundary conditions to estimate branching probabilities (split fractions) in significant accident progression sequences for phenomenological events. Otherwise, USE conservative boundary conditions, as described in Capability Category I.	USE realistic boundary conditions to estimate branching probabilities (split fractions) in all accident sequences for phenomenological events.
PT-B4	ESTIMATE the conditional probability of phenomenologically-induced containment bypass events in a manner that provides an engineering argument that the resulting frequency of containment bypass sequences is larger than would be generated by a realistic analysis.	ESTIMATE the conditional probability of phenomenologically-induced containment bypass events in a realistic manner.	
PT-B5	CHARACTERIZE the probability of containment failure by comparing the magnitude of the containment challenges analyzed in Section 4.4 to the capacities of the affected components and structures analyzed in Section 4.3.		
PT-B6	IDENTIFY accident progression sequences that have the potential for radionuclide release including large release [refer to SR ST-B3].		
PT-B7	ESTIMATE the probability of containment failure events using generic probability evaluations or a treatment of the results of PT-B5 in a manner that provides an engineering argument that the failure probability is larger than would have been generated using a realistic analysis.	ESTIMATE the probability of containment failure events using the results of PT-B5 (for significant accident progression sequences) and generic or conservative probability evaluations for other sequences (as described in Capability Category I).	CALCULATE the probability of containment failure events using the results of PT-B5.

Table 4.5-3 Supporting Requirements for HLR-PT-B (Cont'd)

Index No. PT-B	Capability Category I	Capability Category II	Capability Category III
PT-B8	JUSTIFY the applicability of the generic evaluations used in PT-B7 by comparing their characteristics for physical challenges and structure/component properties with those of the actual as-built, as-operated plant.		
PT-B9	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 (i.e., quantitatively confirm PT-A7.)	ESTIMATE the probability of secondary containment or auxiliary building(s) acting as a retention location preventing or reducing radionuclide release for each accident sequence, accounting for phenomenological effects such as hydrogen burning or external hazards (e.g., seismic event, aircraft crash) [see Notes (13), (14) and (15)].	
PT-B10	CHARACTERIZE the uncertainty range for branching probabilities (split fractions) using engineering judgment.	CHARACTERIZE the uncertainty range for branching probabilities (split fractions) to permit a characterization of uncertainties in applications using structured sensitivity analysis.	CHARACTERIZE the uncertainty distribution of branching probabilities (split fractions) to permit the propagation of uncertainty under PT-E6.

Table 4.5-4 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 developed.

Index No. PT-C	Capability Category I	Capability Category II	Capability Category III
PT-C1	INCLUDE system models that support the accident progression analysis consistent with the applicable requirements of ASME/ANS RA-Sa-2009 technical elements SY (Section 2-2.4) and DA (Section 2-2.6). Also INCLUDE considerations of other hazards (e.g., internal flood, Section 3-2.5; internal fires, Sections 4-2.2, 4-2.3, 4-2.4, 4-2.3; seismic events, Section 5-2.3).		

Table 4.5-4 Supporting Requirements for HLR-PT-C (Cont'd)

Index No. PT-C	Capability Category I	Capability Category II	Capability Category III
PT-C2	USE conservative, generic analyses/evaluations of system success criteria that are applicable to the plant and consistent with the requirement in ASME/ANS RA-Sa-2009 SC-B1, Capability Category I.	USE realistic, generic, or plant-specific analyses for system success criteria for significant accident progression sequences. USE conservative or a combination of conservative and realistic system success criteria for all other accident progression sequences, as described in Capability Category I.	USE realistic, plant-specific system success criteria.
PT-C3	INCLUDE accident sequence dependencies in the logic model for accident progression sequences consistent with the applicable requirements of ASME/ANS RA-Sa-2009 technical element AS (Section 2-2.2), as appropriate for the level of detail of the analysis.		
PT-C4	ASSUME that equipment inside containment does not survive when subjected to environments beyond the equipment's qualification limits. Examples include the following: *(a)* SRV operation at high containment temperature *(b)* vent valve operation at high containment pressure *(c)* motor-operated valve (MOV) operation if located inside containment	INCLUDE adverse environmental impacts on reliability evaluation of equipment inside containment in a realistic manner based on deterministic analyses for the significant accident progression sequences. USE conservative or a combination of conservative and realistic treatments for assessing equipment reliability evaluation inside containment for non-significant accident progression sequences [see examples in Capabilities Category I] [see Note (9)].	INCLUDE adverse environmental impacts on the reliability evaluation of equipment inside containment in a realistic manner based on deterministic analyses [see examples in Capability Category I].

Table 4.5-4 Supporting Requirements for HLR-PT-C (Cont'd)

Index No. PT-C	Capability Category I	Capability Category II	Capability Category III
PT-C5	ASSUME that equipment outside containment does not survive after containment failure if the adverse impacts of containment failure could affect operability, survivability, or alignment of the equipment.	INCLUDE adverse environmental impacts on reliability evaluation of equipment outside containment in a realistic manner given severe accident conditions including containment failure that may be present for the significant accident progression sequences based on probabilistic and/or deterministic analyses. USE conservative or a combination of conservative and realistic treatments for assessing equipment reliability evaluation outside containment for non-significant accident progression sequences with a failed containment [see Note (9)].	INCLUDE containment failure impacts on reliability evaluation of equipment outside containment in a realistic manner based on probabilistic and/or deterministic analyses.
PT-C6	USE a conservative evaluation of secondary side isolation reliability evaluation for significant accident progression sequences caused by SGTR resulting in a large early/large release (if applicable) [see Note (9)].	USE a realistic secondary side isolation reliability evaluation analysis for the significant accident progression sequences caused by SGTR. USE a conservative or a combination of conservative and realistic evaluation of secondary side isolation capability for non-significant accident progression sequences [see Note (9)].	USE a realistic secondary side isolation reliability evaluation analysis for the accident progression sequences caused by SGTR. INCLUDE behavior of relief and isolation valves at applicable temperature and pressure conditions.

Table 4.5-4 Supporting Requirements for HLR-PT-C (Cont'd)

Index No. PT-C	Capability Category I	Capability Category II	Capability Category III
PT-C7 [see Note (10)]	PERFORM a conservative analysis of severe accident-induced SGTR probability (if applicable) that includes plant-specific procedures and design features [see Notes (5) and (9)].	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. PERFORM a realistic or conservative analysis of severe accident-induced SGTR (if applicable) that includes plant-specific procedures and design features that could impact tube failure for non-significant accident progression sequences [see Notes (4) and (9)]. SELECT failure probabilities based on: *(a)* RCS and steam generator post-accident conditions sufficient to describe the important risk outcomes, and *(b)* secondary side conditions including plant-specific treatment of steam generator safety valves and atmospheric dump valves.	PERFORM a realistic analysis of severe accident-induced SGTR (if applicable) that includes plant-specific procedures and design features that could impact tube failure assessment to estimate tube failure probability. USE computer codes that have sufficient capability to estimate the plant-specific conditions that may influence the assessment of SGTR.

Table 4.5-4 Supporting Requirements for HLR-PT-C (Cont'd)

Index No. PT-C	Capability Category I	Capability Category II	Capability Category III
PT-C8 [Note (10)]	PERFORM containment isolation system analysis in a conservative manner. EVALUATE both the failure of containment isolation systems to perform properly and the status of safety systems that do not have automatic isolation provisions [see Note (9)]. INCLUDE consideration of failures of penetrations, seals, and hatches plus pre-existing failures in the containment isolation analyses [see Note (19)].	PERFORM containment isolation system analysis in a realistic manner for the significant accident progression sequences. USE conservative, or a combination of conservative and realistic, treatment for the non-significant accident progression sequences [see Note (9)]. CALCULATE both the failure of containment isolation systems to perform properly and the status of safety systems that do not have automatic isolation provisions. INCLUDE consideration of failures of penetrations, seals, and hatches plus pre-existing failures in the containment isolation analyses [see Note (19)].	PERFORM containment isolation system analysis in a realistic manner. CALCULATE both the failure of containment isolation systems to perform properly and the status of safety systems that do not have automatic isolation provisions. INCLUDE consideration of failures of penetrations, seals, and hatches plus pre-existing failures in the containment isolation analyses [see Note (19)].
PT-C9 [Note (10)]	USE a conservative evaluation of interfacing system failure probability for accident progression sequences resulting in a radionuclide release including a large early / large release [see Note (9)].	PERFORM a realistic interfacing system failure probability analysis for the significant accident progression sequences. USE a conservative or a combination of conservative and realistic evaluation of interfacing system failure probability for non-significant accident progression sequences [see Note (9)]. INCLUDE behavior of piping relief valves, pump seals, and heat exchangers at applicable temperature and pressure conditions.	PERFORM a realistic interfacing system failure probability analysis for the accident progression sequences. INCLUDE behavior of piping, relief valves, pump seals, and heat exchangers at applicable temperature and pressure conditions. PROVIDE static and dynamic failure capabilities, as appropriate.

Table 4.5-4 Supporting Requirements for HLR-PT-C (Cont'd)

Index No. PT-C	Capability Category I	Capability Category II	Capability Category III
PT-C10	ASSUME that operation of mitigating equipment whose injection path to the RPV may be compromised by in-vessel core melt progression, core relocation (e.g., successful flow paths for control rod drive injection into the RPV for BWR) or containment failure is not successful.	QUANTIFY the impact of in-vessel core melt progression, core relocation, and containment failure on continued successful operation of mitigating equipment (e.g., blockage of flow paths for control rod drive hydraulic injection paths into the RPV for BWR).	
PT-C11	ASSUME no fission product scrubbing in the assessment of equipment functionality (perhaps, for example, as a means of reducing local levels of radioactivity).	JUSTIFY any credit taken for fission product scrubbing as a basis for reducing local levels of radioactivity to levels that support equipment functionality (e.g., cite relevant experimental evidence or results of deterministic calculations for the decontamination factor used).	
PT-C12	ASSUME that benevolent failures are not possible [see Note (16)].	QUANTIFY any credit taken for benevolent failures in significant accident progression sequences [see Note (16)].	

Table 4.5-5 Supporting Requirements for HLR-PT-D

Branching probabilities (split fraction) or supporting models for characterizing the reliability of human actions in the accident progression framework shall be developed.

Index No. PT-D	Capability Category I	Capability Category II	Capability Category III
PT-D1	ESTIMATE the probability of human failure events (HFEs) using human action models that support the accident progression analysis consistent with the applicable requirements of ASME/ANS RA-Sa-2009 [1] technical element HR (Section 2-2.5) element; also INCLUDE considerations of other hazards (e.g., internal flood, Sections 3-2.3, 3-2.5; internal fires, Sections 4-2.5, 4-2.10; seismic, Section 5-2.3).		
PT-D2	Conservatively ESTIMATE the probability of HFE following the onset of core damage consistent with applicable procedures, e.g., EOPs/SAMGs, other proceduralized actions, or Technical Support Center guidance [see Note (9)].	Realistically ESTIMATE the probability of HFE following the onset of core damage consistent with applicable procedures, e.g., EOPs/SAMGs, other proceduralized actions, Technical Support Center guidance, and the limitations of HRA methods [see Note (18)].	

Table 4.5-5 Supporting Requirements for HLR-PT-D (Cont'd)

Index No. PT-D	Capability Category I	Capability Category II	Capability Category III
PT-D3	ASSUME no equipment repair (other than AC power recovery) after the onset of core damage.	JUSTIFY credit given for repair (i.e., ensure that plant conditions do not preclude repair; and, ensure actuarial data exists from which to estimate the repair failure probability [see SY-A24, DA-C15 and DA-D8 from ASME/ANS RA-Sa-2009] [1]). AC power recovery based on generic data applicable to the plant is acceptable.	
PT-D4	INCLUDE accident sequence dependencies for operator actions in the logic model for accident progression sequences consistent with the applicable requirements of the ASME/ANS RA-Sa-2009 [1] technical element AS (Section 2-2.2), as appropriate for the level of detail of the analysis.		
PT-D5	ASSUME human actions that are required when human habitability and access are not assured are assigned guaranteed to fail.	INCLUDE environmental impacts on operator actions in a realistic manner based on probabilistic and/or deterministic analyses for the significant accident progression sequences. USE conservative or a combination of conservative and realistic treatment of human actions for non-significant accident progression sequences [see Note (9)].	INCLUDE environmental impacts on human actions in a realistic manner based on probabilistic and/or deterministic analyses.
PT-D6	ASSUME no fission product scrubbing in the assessment of the viability of human actions. For example, assume local levels of radioactivity that may impact human performance are not reduced by scrubbing.	JUSTIFY any credit taken for 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) [see Note (13)].	
PT-D7	ASSUME that benevolent human errors do not occur [see Note (17)].	JUSTIFY any credit taken for benevolent human errors in significant accident progression sequences or in sequences that become non-significant based on the benevolent human error [see Note (17)].	

Table 4.5-6 Supporting Requirements for HLR-PT-E

The frequencies of radionuclide release categories using appropriate models and codes and accounting for method-specific limitations and features shall be calculated.

Index No. PT-E	Capability Category I	Capability Category II	Capability Category III
PT-E1	CALCULATE the RC (as defined in ST-A) frequency consistent with the methods prescribed in PT-A, PT-B, PT-C and PT-D.		
PT-E2	EVALUATE dependencies introduced by common physical parameters involved in multiple CET headers (or equivalent) in a conservative manner [see Notes (1), (2), (3), (6) and (9)].	EVALUATE dependencies introduced by common physical parameters involved in multiple CET headers (or equivalent) in a manner that provides for a realistic estimate of the frequency of significant accident sequences [see Notes (1), (2), (3) and (6)]. EVALUATE dependencies introduced by common physical parameters involved in multiple CET headers (or equivalent) in a manner that provides a conservative or realistic estimate of the frequency of non-significant accident sequences.	EVALUATE dependencies introduced by common physical parameters involved in multiple CET headers (or equivalent) in a realistic manner for all sequences [see Notes (1), (2), (3), and (6)].
PT-E3	EVALUATE the effects of incorporating high failure probability events in the Level 2 PRA logic model on the quantification process [see Note (12)].		
PT-E4	COMPARE the end state frequencies in the Level 2 analysis (e.g., RC frequencies) to the corresponding input frequency from the Level 1 PRA/Level 2 PRA interface (HLR L1) and EXPLAIN the observed differences; INCLUDE: *(a)* the total CDF *(b)* the frequency of each accident sequence end state (i.e., 'accident sequence groups' or PDSs) defined in L1-B5.		
PT-E5	IDENTIFY any limitations in the quantitative results that arise from the effects of high failure probability events.		

Table 4.5-6 Supporting Requirements for HLR-PT-E (Cont'd)

Index No. PT-E	Capability Category I	Capability Category II	Capability Category III
PT-E6	CHARACTERIZE the uncertainty interval for the frequency of RC(s) that represent the largest and earliest releases of radionuclides to the environment. STATE a basis for the estimate consistent with the characterization of parameter uncertainties (see QU-A3, QU-E3, DA-D3, HR-D6, HR-G8, and IE- C15 from the ASME/ANS RA-Sa-2009 [1] and PT-B10).	CHARACTERIZE the frequency uncertainty interval for each RC. ESTIMATE the uncertainty intervals associated with parameter uncertainties, taking into account the state-of-knowledge correlation (see QU-A3, QU-E3, DA-D3, HR-D6, HR-G8, and IE- C15 from ASME/ANS RA-Sa-2009 [1] and PT-B10).	PROPAGATE parameter uncertainties (see DA-D3,HR-D6, HR-G8, and IE-C15 from the ASME/ANS RA-Sa-2009 [1]) using a Monte Carlo approach or other comparable means for each of the radionuclide categories. PROPAGATE parametric uncertainties in such a way that the state-of-knowledge correlation between event probabilities is taken into account from Level 1 PRA analysis through the end of the Level 2 PRA analysis (see QU-A3, QU-E3, DA-D3, HR-D6, HR-G8, and IE-C15 from ASME/ANS RA-Sa-2009 [1] and, PT-B10).
PT-E7	IDENTIFY assumptions made in the development of the PRA model.		
PT-E8	IDENTIFY sources of model uncertainty in the probabilistic treatment of severe accident progression [see Note (15)].		
PT-E9	CHARACTERIZE sources of model uncertainty in the probabilistic treatment of severe accident progression. For example, for each assumption and source of model uncertainty, IDENTIFY how the PRA model is affected (e.g., introduction of a new basic event, changes to basic event probabilities, change in success criterion, changes to radionuclide release frequency, magnitude, or timing, or introduction of a new initiating event).		
PT-E10	DERIVE appropriate truncation limits for accident sequences (or cut sets) to ensure the proper incorporation of frequencies and dependencies in each RC using the requirements of Section 2-2.7 from ASME/ANS RA-Sa-2009 [1] [supporting requirements of ASME/ANS RA-Sa-2009 QU-A3, QU-A4, QU-B1, through QU-B3, QU-B5 through QU-B7 (as appropriate)]. For example, convergence for the significant RCs can be considered sufficient when successive reductions in truncation value of one decade result in decreasing changes, and the final change is less than 5%.		
PT-E11	PERFORM a frequency truncation study to demonstrate the degree of convergence for each RC.		
PT-E12	REVIEW the significant accident progression sequences/cut sets for each RC sufficient to ENSURE that the logic of the cut set or sequence is correct.		
PT-E13	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-E14	REVIEW results for each RC to ENSURE that the flag event settings, mutually exclusive event rules, and recovery rules (if applicable) yield logical results.		

Table 4.5-7 Supporting Requirements for HLR-PT-F

The probabilistic treatment of event progression and source terms consistent with the applicable supporting requirements shall be documented.

Index No. PT-F	Capability Category I	Capability Category II	Capability Category III
PT-F1	DOCUMENT accident progression models and source term calculations in a manner that facilitates PRA applications, upgrades, and peer review.		
PT-F2	DOCUMENT the Level 2 PRA model and source term calculations describing the chronology of events, the accident sequence challenges, containment failure modes, and radionuclide release categories in a manner that facilitates PRA applications, upgrades, and peer review. INCLUDE: *(a)* descriptions of the steps in the significant accident progression sequences from PDS to radiological release *(b)* an explanation of the dominant physical processes, events, and phenomena that contribute to the progressions leading to the resulting outcomes *(c)* a description of equipment environmental challenges and survivability considerations (if equipment is credited beyond equipment qualification).		
PT-F3	DOCUMENT the comparison of severe accident challenges and the containment capacity.		
PT-F4	DOCUMENT the basis for the quantification of split fractions or event probabilities associated with severe accident phenomena.		
PT-F5	DOCUMENT the basis for the HFE and mitigating systems failure probability. INCLUDE a description of: *(a)* the systems credited in the significant accident progression sequences and the justification for their applicability under the conditions in which they are applied *(b)* both the proceduralized (EOP/abnormal operating procedure) and non-proceduralized human actions credited in the significant accident progression sequences and the justification for their applicability under the conditions in which they are applied.		
PT-F6	DOCUMENT the Level 2 PRA quantification and contributors for radionuclide release end states such as the following: *(a)* containment failure modes and phenomena for each significant accident progression sequence *(b)* cut sets or accident progression characteristics that contribute to each significant accident progression sequence *(c)* dominant sequences for each significant RC		
PT-F7	DOCUMENT the truncations used in the model and demonstrate that convergence is adequate.		
PT-F8	DOCUMENT the treatment of high failure probability events in the Level 2 PRA logic model quantification.		
PT-F9	DOCUMENT the characterization of the sources of model uncertainty and assumptions (as identified in PT-E8).		
PT-F10	DOCUMENT the calculated frequency for each RC.		
PT-F11	DOCUMENT the methods used and summarize the results of the importance measures calculated in PT-A9.		