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)

**R. J. Budnitz,** *Cochair*, Lawrence Berkeley National Laboratory
**C. R. Grantom,** *Cochair*, South Texas Project Nuclear Operating Company
**D. W. Henneke,** *Vice Cochair,* General Electric
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**P. J. Amico,** Hughes Associates, Inc.
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**S. A. Bernsen,** Individual
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**J. W. Young,** GE Hitachi
**G. L. Zigler,** Enercon Services

**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:**

**M. T. Leonard,** *Chair 2004 to 2012,* dycoda, LLC

**M. J. Barrett,** Duke Power Company (retired)
**P. Boneham,** Jacobsen Analytics, Ltd.
**D. R. Bradley,** Science Applications International Corporation
**E. T. Burns,** ERIN Engineering & Research, Inc.
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The following project team members were participating at the time that the standard was approved:

**E. T. Burns,** *Chair 2012 - present,* ERIN Engineering & Research, Inc.

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**J. W. Young,** GE Hitachi
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## JCNRM Subcommittee on Risk Application

# 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

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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.<br><br>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 s*ignificant* 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)<br><br>Containment Capacity Analysis (CP)<br><br>Severe Accident Progression Analysis (SA)<br><br>Probabilistic Treatment of Event Progression and Source Terms (PT)<br><br>Radiological Source Term Analysis (ST)<br><br>Evaluation and Presentation of Results (ER)<br><br>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