U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.200, REVISION 3

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Technical Lead: Anders Gilbertson

ACCEPTABILITY OF PROBABILISTIC RISK ASSESSMENT RESULTS FOR RISK-INFORMED ACTIVITIES

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes one approach acceptable to the U.S. Nuclear Regulatory Commission (NRC) staff for determining whether a base probabilistic risk assessment (PRA), in total or in the portions that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors (LWRs). When used in support of an application, this RG will obviate the need for an in-depth review of the base PRA by NRC reviewers, allowing them to focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application.

This revision is consistent with the NRC's PRA Policy Statement, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," dated August 16, 1995 (Ref. 1). This RG endorses, with staff exceptions and clarifications, national consensus PRA standards provided by standards development organizations, and guidance provided by nuclear industry organizations.

Applicability

This RG applies to LWR licensees and applicants subject to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 2), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 3).

The NRC staff may perform a more in-depth review of the PRA acceptability for applications including, but not limited to design certifications (DCs) or combined licenses (COLs), where following aspects of this guide in certain areas of the application may not be feasible or applicable. Other applications and situations may also warrant a more in-depth review including, among others, unique and novel requests, unique technical and factual circumstances, or novel methods and analyses.

Written suggestions regarding this guide or development of new guides may be submitted through the NRC's public Web site in the NRC Library at https://nrc.gov/reading-rm/doc-collections/reg-guides/, under Document Collections, in Regulatory Guides, at https://nrc.gov/reading-rm/doc-collections/reg-guides/contactus.html.

Electronic copies of this RG, previous versions of RGs, and other recently issued guides are also available through the NRC's public Web site in the NRC Library at https://nrc.gov/reading-rm/doc-collections/reg-guides/, under Document Collections, in Regulatory Guides. This RG is also available through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://nrc.gov/reading-rm/adams.html, under ADAMS Accession Number (No.) ML20238B871. The regulatory analysis may be found in ADAMS under Accession No. ML20052C809. The associated draft guide DG-1362 may be found in ADAMS under Accession No. ML19308B636, and the staff responses to the public comments on DG-1362 may be found under ADAMS Accession No. ML20238B873.

Applicable Regulations

- 10 CFR Part 50 provides for the licensing of production and utilization facilities pursuant to the Atomic Energy Act of 1954, as amended and Title II of the Energy Reorganization Act of 1974.

- 10 CFR 50.71, "Maintenance of Records, Making of Reports."

    - 10 CFR 50.71(h)(1) requires that each holder of a COL under Subpart C "Combined Licenses," of 10 CFR Part 52 shall develop a Level 1 and a Level 2 PRA and that the PRA must cover those initiating events and modes for which NRC-endorsed consensus standards on PRA exist 1 year prior to the scheduled date for initial loading of fuel.

    - 10 CFR 50.71(h)(2) requires that each holder of a COL shall maintain and upgrade the PRA required by 10 CFR 50.71(h)(1) and the upgraded PRA must cover initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect 1 year prior to each required upgrade. The PRA must be upgraded every 4 years until the permanent cessation of operations under 10 CFR 52.110(a).

    - 10 CFR 50.71(h)(3) requires that each holder of a COL shall, no later than the date on which the licensee submits an application for a renewed license, upgrade the PRA required by 10 CFR 50.71(h)(1) to cover all modes and all initiating events.

- 10 CFR Part 52 governs the issuance of early site permits, standard DCs, COLs, standard design approvals, and manufacturing licenses for nuclear power facilities pursuant to the Atomic Energy Act of 1954, as amended, and Title II of the Energy Reorganization Act of 1974.

    - 10 CFR 52.47(a)(27) requires that an application under 10 CFR Part 52 contain a final safety analysis report that provides a description of the design-specific PRA and its results.

    - 10 CFR 52.79(a)(46) requires that an application under 10 CFR Part 52 contain a final safety analysis report that provides a description of the plant-specific PRA and its results.

Related Guidance

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 4), provides guidance to the NRC staff in performing safety reviews of construction permit or operating license applications (including requests for amendments) under 10 CFR Part 50 and early site permit, DC, COL, standard design approval, or manufacturing license applications under 10 CFR Part 52 (including requests for amendments).

    - NUREG-0800, Chapter 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors" (Ref. 5), pertains to the staff's review of the design-specific PRA for a DC and the plant-specific PRA for a COL application, respectively.

    - NUREG-0800, Chapter 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 6), is designed to guide the NRC staff in its evaluations of licensee requests for changes to the licensing basis that apply risk insights. Guidance developed in selected application-specific RGs and the corresponding chapters of NUREG-0800 also applies to these types of licensing basis changes.

- NUREG-0800, Chapter 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance" (Ref. 7), addresses the review of risk information used to support permanent plant-specific changes to the licensing basis.

- NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," (Ref. 8), provides guidance on how to treat uncertainties associated with PRA in risk-informed decisionmaking. This guidance is intended to foster an understanding of the uncertainties associated with PRA and their impact on the results of PRA.

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Ref. 9), provides guidance on an acceptable approach for developing risk-informed applications for a licensing basis change that considers engineering issues and applies risk insights.

- RG 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing" (Ref. 10), provides guidance on acceptable methods for using PRA information with established traditional engineering information in the development of risk-informed inservice testing programs.

- RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (Ref. 11), provides guidance on acceptable methods for using risk information to evaluate changes to nuclear power plant technical specification completion times and surveillance frequencies.

- RG 1.178, "An Approach for Plant-Specific, Risk-Informed Decisionmaking for Inservice Inspection of Piping" (Ref. 12), provides guidance on acceptable approaches for meeting the existing requirements in Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, as referenced by 10 CFR 50.55a, "Codes and Standards," for the scope and frequency of inspection of the inservice inspection programs, including the application of risk-informed inservice inspection programs.

- RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance" (Ref. 13), provides guidance on an acceptable method for use in complying with the requirements in 10 CFR 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors," with respect to the categorization of structures, systems, and components (SSCs) that are considered in risk-informing special treatment requirements.

- RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (Ref. 14), provides guidance for use in complying with the requirements for risk-informed, performance-based fire protection programs that comply with 10 CFR 50.48(c), "National Fire Protection Association Standard NFPA 805," and the referenced 2001 Edition of the National Fire Protection Association (NFPA) standard, NFPA 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants" (Ref. 15).

- RG 1.206, "Applications for Nuclear Power Plants" (Ref. 16), provides guidance on the information contained in and submitted with a COL application. This RG discusses the requirements in 10 CFR Part 52 for a COL applicant to conduct a plant-specific PRA and to describe the plant-specific PRA and its results within its final safety analysis report.

- DC/COL-ISG-020, "Interim Staff Guidance on Implementation of a Probabilistic Risk Assessment-Based Seismic Margins Analysis for New Reactors," dated March 15, 2010 (Ref. 17), provides guidance for developing an acceptable PRA-based seismic margins analysis (SMA). In accordance with the Commission direction provided in the Staff Requirements Memorandum (SRM)-SECY-93-087, "SECY-93-087—Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (LWR) Designs," dated July 21, 1993 (Ref. 18), DC and COL applicants may perform a PRA-based SMA to assess seismic risk.

- DC/COL-ISG-028, "Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application," issued November 2016 (Ref. 19), provides guidance to DC and COL applicants that supplements ASME/American Nuclear Society (ANS) PRA standard ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Ref. 20). The guidance in DC/COL-ISG-028 is needed because ASME/ANS RA-Sa-2009 was developed for currently operating reactors. As a result, for PRAs developed for the DC and COL application stages, some supporting requirements (SRs) in ASME/ANS RA-Sa-2009 are not applicable or cannot be achieved as written, while other SRs need some clarification to understand how they can be achieved.

Purpose of Regulatory Guides

The NRC issues RGs to describe to the public methods that the staff considers acceptable for use in implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific problems or postulated events, and to provide guidance to applicants. RGs are not substitutes for regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs will be deemed acceptable if they provide a sufficient basis for the findings required for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), approval numbers 3150-0011 and 3150-0151. Send comments regarding this information collection to the Information Services Branch (T6-A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0011 and 3150-0151), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW Washington, DC 20503; e-mail: oira_submission@omb.eop.gov.

Public Protection Notification

The NRC may not conduct or sponsor, and a person is not required to respond to, a collection of information unless the document requesting or requiring the collection displays a currently valid OMB control number.

B. DISCUSSION

Reason for Revision

This revision of RG 1.200 (Revision 3) addresses new industry guidance and issues identified since the last revision was issued. Specifically, this revision accomplishes the following:

- endorses Nuclear Energy Institute (NEI) 17-07, Revision 2, "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard," issued August 2019 (Ref. 21) (see regulatory positions C.2.2.2 through C.2.2.4)

- endorses, with staff exceptions and clarifications, requirements in ASME/ANS RA-S Case 1, "Case for ASME/ANS RA-Sb-2013 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment of Nuclear Power Plant Applications," dated November 22, 2017 (Ref. 22) (see Appendix B)

- endorses the following portions of the document from the Pressurized Water Reactor Owners Group (PWROG), PWROG-19027-NP, Revision 2, "Newly Developed Method Requirements and Peer Review," issued July 2020 (Ref. 23):

  - requirements for the peer review of newly developed methods (NDMs) (see regulatory positions C.2.2.2 through C.2.2.4)

  - process for determining whether a change to a PRA is classified as PRA maintenance or a PRA upgrade (see Appendix C)

  - definitions related to NDMs, PRA maintenance, and PRA upgrade (see Glossary)

- enhances guidance related to key assumptions and sources of uncertainty (see regulatory position C.3.3.2)

- provides a glossary of key terms

- provides a list of hazards to be considered in the development and use of PRA (see Appendix D)

The staff's previous endorsement of ASME/ANS RA-Sa-2009 in RG 1.200, Revision 2, issued March 2009, (Ref. 24), has been modified to reflect endorsement of terms and their definitions derived from PWROG-19027-NP, Revision 2. However, all other portions of the staff's endorsement of ASME/ANS RA-Sa-2009 from RG 1.200, Revision 2, are otherwise not changed by this revision, as provided in Appendix A to this RG.

Background

In 1995, the NRC issued a policy statement titled, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement" (Ref. 1). This policy statement encourages the use of PRA in all regulatory matters and states that, "the use of PRA technology should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach." Additionally, on July 28, 2000, the staff issued SECY-00-0162, "Addressing PRA Quality in Risk-Informed Activities" (Ref. 25), which describes an approach for addressing PRA quality in risk-informed activities, including identification of the scope and minimal functional attributes of a technically acceptable PRA. Subsequently, on July 13, 2004, the staff issued SECY-04-0118, “Plan for the Implementation of the Commission’s Phased Approach to Probabilistic Risk Assessment Quality” (Ref. 26), which presents the staff’s approach to defining the needed PRA quality for current or anticipated applications, as well as the process for achieving this quality, while allowing risk-informed decisions to be made using currently available methods until all of the necessary guidance documents are developed and implemented. SECY-07-0042, “Status of the Plan for the Implementation of the Commission’s Phased Approach to Probabilistic Risk Assessment Quality,” dated March 7, 2007 (Ref. 27), provides an update to the staff’s plan.

Since issuance of the 1995 NRC policy statement, many applications of PRA have been implemented or undertaken in risk-informed regulatory activities, including modification of the NRC’s reactor safety inspection program and initiation of work to modify reactor safety regulations.

Fundamentally, the staff must have confidence that the information developed from a PRA is sound and reliable. Consequently, the PRA’s technical content needs to be complete, correct, and accurate and produce insights with appropriate fidelity to support any decision contemplated. As a result, the sufficiency of a PRA’s technical content determines the acceptability of a PRA.[1] PRA acceptability describes the ability of a PRA to support risk-informed regulatory decisionmaking and, for a base PRA, is defined in terms of the NRC regulatory position in Section C of this RG, national consensus PRA standards requirements, and peer review processes. These three aspects each depend on the other to achieve PRA acceptability, as illustrated in Figure 1.

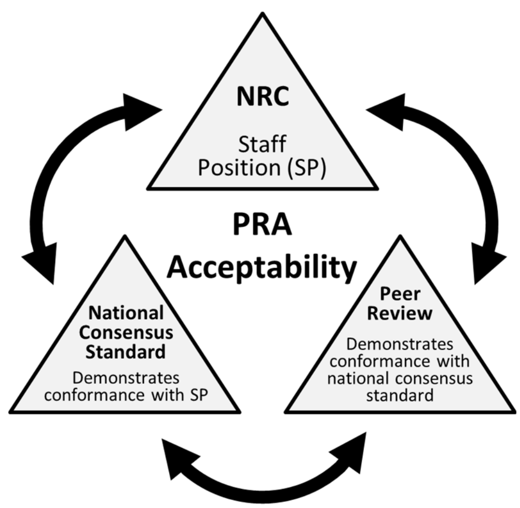

Figure 1. NRC general framework for achieving PRA acceptability

National consensus PRA standards provide one set of minimum requirements that can be met, as endorsed by the staff with exceptions and clarifications, for a base PRA to be considered acceptable. Because these PRA standards use the terms “requirement,” “require,” and other similar mandatory language, the staff’s endorsement, including staff exceptions and clarifications, mirrors this language.

1 The term “PRA acceptability” and related phrasings are synonymous with previously used terms such as “PRA quality” and “PRA technical adequacy.” The staff is using the term “PRA acceptability” with respect to the scope, level of detail, conformance with PRA technical elements (i.e., technical adequacy), and plant representation of a PRA as related to the outcome of the NRC staff’s review of a given risk-informed application. For additional information, see DPO-2016-001 (ADAMS Accession No. ML17013A015).

However, the use of this language in this RG is not intended to convey a regulatory requirement or suggest that these standards are the only way to meet the statutory and regulatory requirements.

National consensus PRA standards include both technical requirements and process-related requirements such as those related to peer review and PRA configuration control. The PRA peer review process is used to determine whether a base PRA meets the requirements provided in the national consensus PRA standard.[2] One acceptable approach for performing a peer review of a PRA is to perform an established, NRC-endorsed peer review process by qualified personnel that documents the results and identifies both strengths and weaknesses of the base PRA. When used in support of an application, the use of this RG will obviate the need for an in-depth review of the base PRA by NRC reviewers, allowing them to focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application. The acceptability of a base PRA is measured against the base PRA scope, level of detail, conformance with the NRC regulatory position in Section C of this RG, and representation of the modeled plant.

This RG provides guidance to licensees and applicants on how to meet the regulatory positions in Section C for determining the acceptability of the base PRA used in support of a risk-informed regulatory activity. It also endorses, with staff exceptions and clarifications, the national consensus PRA standards and industry guidance on how to perform a PRA peer review process.

National consensus PRA standards have been developed by ASME and ANS. In February 2009, ASME and ANS jointly issued ASME/ANS RA-Sa-2009, referred to hereafter as the ASME/ANS Level 1/LERF PRA standard. The ASME/ANS Level 1/LERF PRA standard is for an at-power Level 1 and limited Level 2 PRA of internal and external hazards for LWRs. Appendix A to this RG provides the staff endorsement, with exceptions and clarifications, of the requirements in ASME/ANS RA-Sa-2009. Other than the staff's endorsement of terms and definitions in PWROG-19027-NP, Revision 2, the staff's endorsement from RG 1.200, Revision 2 is otherwise not changed, as provided in Appendix A in this RG. The staff's endorsement, with exceptions and clarifications, is based on the staff's review of the requirements in the ASME/ANS Level 1/LERF PRA standard against the related regulatory position in Section C of this RG.

Additionally, in November 2017, ASME and ANS jointly issued ASME/ANS RA-S Case 1, which provides a proposed alternative set of requirements related to the requirements for seismic PRA in Part 5 of ASME/ANS RA-Sa-2009. Appendix B to this RG provides the staff endorsement, with exceptions and clarifications, of the requirements for seismic PRA in ASME/ANS RA-S Case 1. Similar to the staff endorsement of the ASME/ANS Level 1/LERF PRA standard, the staff endorsement, with exceptions and clarifications, of ASME/ANS RA-S Case 1 is based on the staff's review of the seismic PRA requirements in ASME/ANS RA-S Case 1 against the related regulatory position in Section C of this RG.

A PRA peer review process was developed and has been applied by reactor owners' groups and other industry organizations for several years. NEI issued the PRA peer review guidance document NEI 17-07, Revision 2, which provides guidance on how to perform a PRA peer review to meet the PRA peer review requirements in the ASME/ANS Level 1/LERF PRA standard. Consistent with the scope of the ASME/ANS Level 1/LERF PRA standard, NEI 17-07, Revision 2, addresses PRA peer reviews for internal and external hazards as well as follow-on PRA peer reviews. Regulatory position C.2.2 of this RG provides guidance on the performance of PRA peer reviews and endorses NEI 17-07, Revision 2, in its entirety as a means of satisfying the peer review requirements in the ASME/ANS Level 1/LERF PRA

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2 DC/COL-ISG-028 provides guidance on the use of peer reviews and self-assessments to establish the acceptability of PRAs performed to support DC and COL applications.

standard, as endorsed by the NRC in this RG with staff exceptions and clarifications. The following are other related NEI guidance documents on PRA peer review from which NEI 17-07, Revision 2, was developed:

- NEI 00-02, Revision A3, "Probabilistic Risk Assessment Peer Review Process Guidance," dated March 20, 2020 (Ref. 28)

- NEI 05-04, Revision 2, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," issued November 2008 (Ref. 29)

- NEI 07-12, Revision 0, Draft H, "Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines," issued November 2008 (Ref. 30)

- NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," issued August 2012 (Ref. 31), as accepted by the NRC by letter dated March 7, 2018 (Ref. 32)

- NEI Appendix X, "NEI 05-04/07-12/12-06, Appendix X: 'Close-Out of Facts and Observations (F&Os)'," dated February 21, 2017 (Ref. 33), as accepted by the NRC by letter dated May 3, 2017 (Ref. 34)

This revision provides guidance on one way to determine the acceptability needed for a base PRA that is used in a risk-informed integrated decisionmaking process. More specifically, this RG provides guidance in the following four areas:

1. defining the acceptability of a base PRA

2. the NRC's position on national consensus PRA standards, industry PRA peer review process documents, and other related industry documents

3. demonstration that the base PRA used in regulatory applications is acceptable, in total or in the parts that are used to support an application

4. documentation to support a regulatory decision

This RG does not address non-PRA approaches such as bounding analyses; rather, this guide only addresses PRA approaches. Instead, NUREG-1855 provides guidance on how to perform acceptable bounding analyses and on limiting the scope of the application. RGs addressing specific applications, such as RG 1.201, allow for the use of PRAs that are not full scope (e.g., they may not include contributions from external initiating events or low-power and shutdown (LPSD) modes of operation). Those RGs do, however, state that the missing PRA scope items are to be addressed in some way, such as using bounding analyses to justify excluding missing PRA scope items, or by limiting the scope of the application.

This RG is a supporting document to other NRC RGs that address risk-informed activities. Other application-specific RGs provide guidance on how the base PRA can be used in the decision under consideration. As such, other RGs refer to RG 1.200 in order to ensure the acceptability of the base PRA from which the application-specific PRA is derived. PRA acceptability for a given risk-informed activity is determined considering the staff positions in this RG, staff positions in relevant application-specific regulatory guidance, and any related requirements (e.g., license conditions) for the application.

Examples of application-specific regulatory guidance documents include (1) RG 1.174 and NUREG-0800, Chapter 19.2, which provide general guidance on applications that address changes to the licensing basis (2) application-specific RGs, such as RG 1.175, RG 1.177, and RG 1.178, and (3) RGs associated with implementation of certain regulations, such as RG 1.201, particularly those that rely on a plant-specific PRA to implement the rule (e.g., 10 CFR Part 52). Figure 2 illustrates the relationship of RG 1.200 to some examples of risk-informed activities, application-specific guidance (e.g., RG 1.174), national consensus PRA standards, and industry programs.

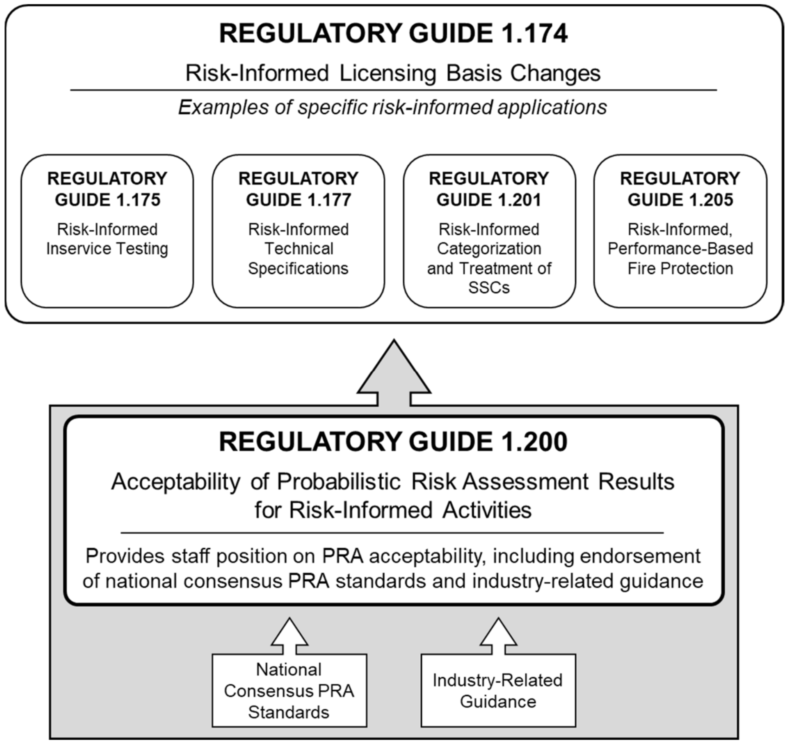

Figure 2. Relationship of RG 1.200 to other risk-informed guidance

Application-specific RGs have been and continue to be developed to provide guidance on the use of PRA information. For example, RG 1.174 and the related guidance in Chapter 19.2 of NUREG-0800 provide general guidance on applications that address risk-informed changes to the licensing basis. Although RG 1.174 is written in the context of one regulatory activity specific to operating reactors (license amendments), its underlying philosophy and principles are applicable to a broad spectrum of reactor regulatory activities that use RG 1.200 for determining the acceptability of the base PRA. Key aspects of RG 1.174 and Chapter 19.2 of NUREG-0800 include the following:

- Description of a risk-informed integrated decisionmaking process that characterizes how risk information is used and, more specifically, clarifies that such information is one element of the decision-making process. That is, decisions “are expected to be reached in an integrated fashion, considering traditional engineering and risk information, and may be based on qualitative factors as well as quantitative analyses and information.”

- Recognition that the PRA developed to support regulatory decisions (i.e., as derived from the base PRA) can vary in terms of the scope, level of detail, and the level of conformance with the technical elements in a PRA standard. The PRA is to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decisionmaking process.

For some applications and decisions, only specific portions of a base PRA may need to be used. In other applications, the full-scope base PRA may need to be used. General guidance regarding the appropriate scope, requirements to be met in national consensus PRA standards, level of detail, and plant representation for a PRA used in a specific application is provided in the related application-specific guidance documents for those activities.

Consideration of International Standards

The International Atomic Energy Agency (IAEA) works with member states and other partners to promote the safe, secure, and peaceful use of nuclear technologies. The IAEA develops Safety Requirements and Safety Guides for protecting people and the environment from harmful effects of ionizing radiation. This system of safety fundamentals, safety requirements, safety guides, and other relevant reports, reflects an international perspective on what constitutes a high level of safety. To inform its development of this RG, the NRC considered, for review and incorporation if appropriate, the IAEA Safety Requirements and Safety Guides[3] pursuant to the Commission's International Policy Statement and Management Directive and Handbook 6.6 (Ref. 35).

The following IAEA Safety Standards Series incorporate similar design and preoperational testing guidelines and are consistent with the basic safety principles considered in developing this RG:

- IAEA Safety Standards Series No. SSG-3, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants," issued 2010 (Ref. 35)

- IAEA Safety Standards Series No. SSG-4, "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants," issued 2010 (Ref. 37)

Documents Discussed in Staff Regulatory Guidance

This RG endorses, with exceptions and clarifications, the use of one or more codes or standards developed by external organizations, and other third-party guidance documents. These codes, standards and third-party guidance documents may contain references to other codes, standards, or third-party guidance documents ("secondary references"). If a secondary reference has itself been incorporated by reference into NRC regulations as a requirement, then licensees and applicants must comply with that standard as set forth in the regulation. If the secondary reference has been endorsed in a RG as an acceptable approach for meeting an NRC requirement, then the standard constitutes a method acceptable to the NRC staff for meeting that regulatory requirement as described in the specific RG. If the secondary reference has neither been incorporated by reference into NRC regulations nor endorsed in a RG, then the secondary reference is neither a legally-binding requirement nor a "generic" NRC approved acceptable approach for meeting an NRC requirement. However, licensees and applicants may consider and use the information in the secondary reference, if appropriately justified, consistent with current regulatory practice, and consistent with applicable NRC requirements.

3 Such information related to this guide may be found at WWW.IAEA.Org/ or by writing the International Atomic Energy Agency, P.O. Box 100 Wagramer Strasse 5, A 1400 Vienna, Austria; telephone (+431) 2600-0; fax (+431) 2600-7; or e mail Official.Mail@IAEA.Org. It should be noted that the requirements specified in the NRC's regulations take precedence over any non-corresponding international recommendations.

C. STAFF REGULATORY GUIDANCE

C.1. An Acceptable Base Probabilistic Risk Assessment

This section describes one acceptable approach for defining the acceptability of a base PRA used in regulatory decision-making for commercial LWR nuclear power plants. As defined in the Glossary of this RG, an approach is considered to be a PRA when it (1) provides a quantitative assessment of the identified risk in terms of scenarios that result in undesired consequences (e.g., core damage or a large early release) and their frequencies and (2) comprises specific technical elements in performing the quantification.

The base PRA is defined as the PRA from which results or insights are derived or that is modified or manipulated to support a risk-informed NRC regulatory activity. The base PRA provides a quantitative assessment of the identified risk of the as-built and as-operated plant in terms of scenarios that result in undesired consequences (e.g., core damage or a large early release) and their frequencies and comprises specific technical elements in performing the quantification. The base PRA serves as the foundational representation of the as-built and as-operated plant necessary to support an application. In some cases, such as applications related to 10 CFR 50.69, the PRA used in the application may be the base PRA.

Regulatory position C.1 of this RG and its subsections provide guidance in the following four areas that collectively determine the acceptability of a base PRA:

- Scope of a base PRA: The scope of a base PRA is defined in terms of (1) the metrics used to characterize risk; (2) the plant operating states (POSs) for which the risk is to be evaluated; and (3) the causes of initiating events (hazard groups) that can potentially challenge and disrupt the normal operation of the plant and, if not prevented or mitigated, would eventually result in core damage or a large release, or both. The scope of a base PRA is determined by its intended use for representing the as-built and as-operated plant. Regulatory position C.1.1 provides guidance with respect to full-scope Level 1 and Level 2 PRAs.

- Technical elements of a base PRA: The PRA technical elements are defined in terms of the fundamental technical analyses needed to develop and quantify the base PRA model for its intended purpose (e.g., determination of a specific risk metric). The characteristics and attributes of the PRA technical elements define specific requirements that should be met to successfully perform those technical analyses and achieve a defined objective. Regulatory position C.1.2 provides guidance on the technical elements of full-scope Level 1 and Level 2 PRAs.

- Level of detail of a base PRA: The level of detail of a base PRA is defined in terms of the resolution of the modeling used to represent the behavior and operations of the plant. A minimal level of detail is necessary to ensure that the impacts of designed-in dependencies (e.g., support system dependencies, functional dependencies, and dependencies on operator actions) are correctly captured. This minimal level of detail is implicit in the technical elements comprising the base PRA and their associated characteristics and attributes. Regulatory position C.1.3 provides guidance on the level of detail for a base PRA.

- Plant representation and PRA configuration control: Plant representation is defined in terms of how closely the base PRA represents the plant as it is actually built and operated. In general, PRA results used to support an application must be derived from a base PRA model that represents the as-built and as-operated plant to the extent needed to support the application. Consequently, the base PRA is maintained and upgraded, where necessary, to ensure it represents the as-built and as-operated plant. However, for some applications, the plant may only be in the DC or COL stage of licensing or may be under construction, at which points the plant may not have been built or is not yet operational. At these licensing stages, the base PRA model is intended to reflect the as-designed plant. Regulatory position C.1.4 provides guidance on plant representation in the base PRA.

C.1.1. Scope of a Base Probabilistic Risk Assessment

The scope of a base PRA is defined by the challenges included in the analysis, the level of analysis performed, and its intended use for representing the as-built and as-operated plant. Specifically, the base PRA scope is defined in the following terms:

- metrics used to characterize the risk,
- POSs for which the risk is to be evaluated, and
- causes of initiating events (hazard groups) that can potentially challenge and disrupt the normal operation of the plant.

For currently operating reactors and for reactors at the DC or COL application stage, some applications may require a full-scope Level 1 PRA and some aspects of a Level 2 PRA.

Risk characterization is typically expressed by metrics of core damage frequency (CDF) and large early release frequency (LERF) (as surrogates for latent and early fatality risks, respectively, for operating LWRs). Large release frequency (LRF) is used as a risk metric for LWR DC and COL applicants, as approved in SRM-SECY-90-16, "SECY-90-16—Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," dated June 26, 1990 (Ref. 38). The CDF and LERF metrics are defined in a functional sense as follows:

- **Core damage frequency (CDF)** is defined as the sum of the frequencies of those accidents that result in 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.
- **Large early release frequency (LERF)** is defined as the sum of the frequencies of those accidents leading to rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions such that there is the potential for early health effects. (Such accidents generally include unscrubbed releases associated with early containment failure shortly after vessel breach, containment bypass events, and loss of containment isolation.)

As discussed in SECY-13-0029, "History of the Use and Consideration of the Large Release Frequency Metric by the U.S. Nuclear Regulatory Commission," dated March 22, 2013 (Ref. 39), the staff has not developed a definition of LRF. The staff encourages DC and COL applicants to review approved DC and COL applications (including the associated staff safety evaluation reports) when developing a definition of LRF. In accordance with SRM-SECY-12-0081, "Staff Requirements—SECY-12-0081—Risk-Informed Regulatory Framework for New Reactors," dated October 22, 2012 (Ref. 40), COL holders transition from the use of LRF to the use of LERF when the fuel-load PRA required by 10 CFR 50.71(h)(1) is developed.

Issues related to the reliability of barriers (in particular, containment integrity and consequence mitigation) are addressed by other parts of the decisionmaking process, such as consideration of defense in depth. To provide the risk perspective for use in decision-making, a Level 1 PRA assesses the CDF risk metric and a limited Level 2 PRA assesses the LERF risk metric.

DC and COL applicants should meet certain requirements in the most recent NRC-endorsed ASME/ANS Level 1/LERF PRA standard, ASME/ANS RA-Sa-2009 (Ref. 20), such as information relating to spatial aspects of SSCs if that information is available. Also, COL holders should meet certain requirements in ASME/ANS RA-Sa-2009 when that information can be obtained (e.g., requirements to perform walkdowns of the as-built plant, requirements to use plant-specific operating experience when plant-specific experience becomes available after operating for some time).

Plant operating states (POSs) are used to subdivide the plant operating cycle into unique states, such that the plant response can be assumed to be the same within the given POS for a given initiating event. Operational characteristics (such as reactor power level; in-vessel temperature, pressure, and coolant level; equipment operability; and changes in decay heat load or plant conditions that allow new success criteria or reactor coolant system or containment configuration) are examined to identify those relevant to defining POSs. These characteristics are used to define the states, and the fraction of time spent in each state is estimated using plant-specific information. The risk perspective is based on the total risk associated with the operation of the reactor, which includes not only at-power operation but also LPSD conditions. For some applications, the risk impact may affect some modes of operation but not others.

A hazard group is a group of similar hazards that are assessed in a PRA using common approaches, methods, and likelihood data for characterizing the effect on the plant. A hazard is a category of similar challenges to plant operations that poses some risk to a facility. For example, internal events is a hazard group, whereas loss-of-coolant accidents (LOCAs) are a hazard within the internal events hazard group. A hazard group is characterized as either an internal or external hazard type, where the distinction between these hazard types is defined by the plant boundary in the PRA. The hazard groups addressed in this RG include the following:

- internal events
- internal flood
- internal fire
- seismic events
- high wind
- external flood
- other hazards

The first six hazard groups listed represent categories of hazards that are typically analyzed and modeled in detail using a PRA. However, a key feature of a base PRA is that a wide spectrum of potential hazards in terms of magnitude and frequency of occurrence should be systematically surveyed to help ensure that significant contributors to plant risk are not inadvertently excluded from the PRA. As such, there are a number of internal and external hazards that are considered during the development of a base PRA in addition to those hazards analyzed under the first six hazard groups listed above. For many such internal and external hazards, the risk posed to a facility can be assessed qualitatively, quantitatively, or both, but in a simplified manner and without the need for a detailed PRA model. Regulatory position C.1.2.6 provides additional guidance on screening and conservative analyses that can be performed to this end. Conversely, some such internal and external hazards may produce impacts to a plant and a potential plant response that are too complex for a simplified analysis and should be modeled in detail using a PRA. This latter type of hazard is commonly referred to as an “other hazard.” Regulatory position C.1.2.9 provides additional guidance on the modeling of such hazards. Appendix D to this RG provides a listing of and general description for the internal and external hazards that should be considered during the development of a base PRA.

Initiating events are perturbations to the steady state operation of the plant that challenge plant control and safety systems and could lead to core damage, radioactivity release, or both. They also include failures of plant control and safety systems that may cause perturbation to the steady-state operation of the plant that could lead to these same outcomes. Initiating events may be caused by internal hazards such as equipment failure, operator actions, or a flood or fire internal to the plant, or by external hazards such as an earthquake, external flood, or high wind. The risk perspective is based on a consideration of total risk, which includes risk contributions from both internal and external hazards.

C.1.2. Technical Elements of a Base Probabilistic Risk Assessment and Associated Characteristics and Attributes

The PRA technical elements are defined in terms of the fundamental technical analyses needed to develop and quantify the base PRA model for its intended purpose (e.g., determination of a specific risk metric). The characteristics and attributes of the PRA technical elements define specific requirements that should be met to successfully perform those technical analyses and achieve a defined objective.

Table 1 provides the list of general technical elements that are necessary for acceptable Level 1 and Level 2 base PRAs. A base PRA that is missing one or more of these elements would not be considered a complete base PRA.

Table 1. Technical Elements of a Base PRA

Scope of Analysis	Technical Element	
Level 1	• Initiating event analysis • Success criteria analysis • Accident sequence analysis • Systems analysis	• Parameter estimation analysis • Human reliability analysis • Quantification
Level 2	• Plant damage state analysis • Severe accident progression analysis	• Quantification • Source term analysis
Interpretation of results and documentation are technical elements of Level 1 and Level 2 base PRAs.		

These technical elements are applicable to the base PRA models constructed to address each of the risk contributors (hazard groups) for each of the POSs. Because additional analyses are required to characterize their impact on the plant in terms of causing initiating events and mitigating equipment failures, regulatory positions C.1.2.3, C.1.2.4, C.1.2.5, C.1.2.7, and C.1.2.8 discuss internal flood, internal fire, seismic, high wind and external flood PRAs, respectively. As it may be possible to screen some hazards, regulatory position C.1.2.6 covers screening and conservative analyses and regulatory position C.1.2.9 provides a discussion of the technical elements associated with PRA for other hazards. While the technical elements are the same for each POS, within a specific technical element, other considerations may need to be addressed for LPSD conditions, which are discussed in regulatory position C.1.2.10. In order to better understand the results, it is important to examine the different contributors on both individual and relative bases. Therefore, the technical element on the interpretation of results is discussed separately in regulatory position C.1.2.11. The technical element on documentation of the PRA is discussed in regulatory position C.1.2.12.

C.1.2.1 Technical Elements for a Level 1, Internal Events, At-Power Probabilistic Risk Assessment

This section identifies the technical elements for a Level 1, internal events, at-power PRA. The objective for each technical element is briefly described, and the characteristics and attributes needed to achieve the objective are provided in Table 2. The technical elements for a Level 1, internal events, at-power PRA are the following:

- initiating event analysis,
- success criteria analysis,
- accident sequence analysis,
- systems analysis,
- parameter estimation analysis,
- human reliability analysis, and
- quantification.

Initiating event analysis identifies and characterizes the events that both challenge normal plant operation during power or shutdown conditions and require successful mitigation by plant equipment and personnel to prevent core damage from occurring. Events that have occurred at the plant and those that have a reasonable probability of occurring are identified and characterized. An understanding of the nature of the events is performed such that a grouping of the events, with the groups defined by similarity of system and plant responses (based on the success criteria), may be performed to manage the large number of potential events that can challenge the plant.

Success criteria analysis determines the minimum requirements for each function (and ultimately the systems used to perform the functions) to prevent core damage (or to mitigate a release) given an initiating event. The requirements defining the success criteria are based on acceptable engineering analyses that represent the design and operation of the plant under consideration. For a function to be successful, the criteria are dependent on the initiator and the conditions created by the initiator. The computer codes used to perform the analyses for developing the success criteria are validated and verified for both technical integrity and suitability to assess plant conditions for the reactor pressure, temperature, and flow range of interest, and they accurately analyze the phenomena of interest. Calculations are performed by personnel who are qualified to perform the types of analyses of interest and are well trained in the use of the codes.

Accident sequence analysis models, chronologically (to the extent practical), the different possible progressions of events (i.e., accident sequences) that can occur from the start of the initiating event to either successful mitigation or core damage. The accident sequences account for the systems that are used (and available) and operator actions performed to mitigate the initiator based on the defined success criteria and plant operating procedures (e.g., plant emergency and abnormal operating procedures) and training. The availability of a system includes consideration of the functional, phenomenological, and operational dependencies and interfaces between the various systems and operator actions during the course of the accident progression.

Systems analysis identifies the various combinations of failures that can prevent the system from performing its function as defined by the success criteria. The model representing the various failure combinations includes, from an as-built and as-operated perspective, the system hardware and instrumentation (and their associated failure modes) and human failure events (HFEs) that would prevent the system from performing its defined function. Human-induced security events (e.g., sabotage, malevolent acts) are not included in the scope of considered HFEs. The basic events representing equipment and HFEs are developed in sufficient detail in the model to account for dependencies among the various systems and to distinguish the specific equipment or human events that have a major impact on the system's ability to perform its function.

Parameter estimation analysis quantifies the frequencies of the initiating events, as well as the equipment failure probabilities and equipment unavailabilities of the modeled systems. The estimation process includes a mechanism for addressing uncertainties and has the ability to combine different sources of data in a coherent manner, including the actual operating history and experience of the plant when it is of sufficient quality, as well as applicable generic experience.

Human reliability analysis identifies and provides probabilities for the HFEs that can negatively impact normal or emergency plant operation. The HFEs associated with normal plant operation include the events that leave the system (as defined by the success criteria) in an unrevealed, unavailable state. The HFEs associated with emergency plant operation represent those human actions that, if not performed, do not allow the needed system to function. Quantification of the probabilities of these HFEs is based on plant- and accident-specific conditions, where applicable, including any dependencies among actions and conditions.

Quantification provides an estimation of the CDF given the design or operation of the plant (depending whether the plant is in the design or operating stage). Regardless of the plant stage, the CDF is based on the summation of the estimated CDF from each accident sequence for each initiator group. If truncation of accident sequences and cut sets is applied, truncation limits are set so that the overall model results are not impacted in such a way that significant accident sequences or contributors are eliminated. Therefore, the truncation value is selected so that the required results are stable with respect to further reduction in the truncation value.

Table 2 provides a summary of characteristics and attributes needed for the technical elements for a Level 1 PRA for internal events. The characteristics and attributes are provided for at-power conditions.

Table 2. Summary of Technical Characteristics and Attributes of a Level 1, Internal Events PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
Initiating Event Analysis	• Sufficiently detailed identification and characterization of initiating events • Grouping of individual events according to plant response and mitigating requirements • Proper screening of any individual or grouped initiating events Note: It is recognized that for those new reactor designs with substantially lower risk profiles (e.g., internal events CDF below 1×10^{-6} per year) that the quantitative screening value may need to be adjusted according to the corresponding baseline risk value.
Success Criteria Analysis	• Based on best estimate engineering analyses applicable to the actual plant design and operation, as available • Codes developed in sufficient detail to— – analyze the phenomena of interest – be applicable in the pressure, temperature, and flow range of interest
Accident Sequence Development Analysis	• Defined in terms of hardware, operator action, and timing requirements and desired end states (e.g., core damage or plant damage states) • Includes necessary and sufficient equipment (safety- and non-safety-related) reasonably expected to be used to mitigate initiators • Includes functional, phenomenological, and operational dependencies and interfaces

Table 2. Summary of Technical Characteristics and Attributes of a Level 1, Internal Events PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
Systems Analysis	Models developed in sufficient detail to achieve the following purposes: • Reflect the as-designed, as-built, and as-operated plant (as applicable), including how it has performed during the plant history for operating plants • Reflect the success criteria for the systems to mitigate each identified accident sequence • Capture the impact of dependencies, including support systems and abnormal environmental impacts • Include both active and passive components and failure modes that impact the function of the system • Include common-cause failures, human errors, unavailability resulting from test and maintenance, etc.
Parameter Estimation Analysis	• Estimation of parameters associated with initiating events, basic event probability models, recovery actions, and unavailability events using plant-specific and generic data as applicable • Estimation is consistent with component boundaries • Estimation includes a characterization of the uncertainty
Human Reliability Analysis	• Identification and definition of the HFEs that would result in initiating events or pre- and post-accident HFEs that would impact the mitigation of initiating events • Quantification of the associated human error probabilities (HEPs) considering scenario- (where applicable) and plant-specific factors (as available) and including appropriate dependencies (both pre- and post-accident)
Quantification	• Estimation of the CDF for modeled sequences that are not screened as a result of truncation, given as a mean value • Estimation of the accident sequence CDFs for each initiating event group • Truncation values set relative to the total plant CDF such that the CDF is stable with respect to further reduction in the truncation value

NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," issued April 2005 (Ref. 41), and NUREG-1842, "Evaluation of Human Reliability Analysis Methods Against Good Practices," issued September 2006 (Ref. 42), provide good practices for meeting the above characteristics and attributes for human reliability analysis.

C.1.2.2 Technical Elements for a Level 2, Internal Events, At-Power Probabilistic Risk Assessment

This section identifies the technical elements for a Level 2, internal events, at-power PRA. The objective for each technical element is briefly described and the characteristics and attributes needed to achieve the objective are provided in Table 3. The technical elements for a Level 2, internal events, at-power PRA are the following:

- plant damage state analysis,
- severe accident progression analysis,
- source term analysis, and
- quantification.

Plant damage state analysis groups similar core damage scenarios together to allow a practical assessment of the severe accident progression and containment response resulting from the full spectrum of core damage accidents identified in the Level 1 analysis. The plant damage state analysis defines the attributes of the core damage scenarios that represent boundary conditions in the assessment of severe accident progression and containment response that ultimately affect the resulting radionuclide releases. The attributes address the dependencies between the containment systems modeled in the Level 2 analysis with the core damage accident sequence models to fully account for mutual dependencies. Core damage scenarios with similar attributes are grouped together to allow for efficient evaluation of the Level 2 response.

Severe accident progression analysis models the different series of events that challenge containment integrity for the core damage scenarios represented in the plant damage states. The accident progressions account for interactions among severe accident phenomena and system and human responses to identify credible containment failure modes, including failure to isolate the containment. The timing of major accident events and the subsequent loadings produced on the containment are evaluated against the capacity of the containment to withstand the potential challenges. The containment performance during the severe accident is characterized by the timing (e.g., early versus late), size (e.g., catastrophic versus bypass), and location of any containment failures. The codes used to perform the analysis are validated and verified for both technical integrity and suitability. Calculations are performed by personnel qualified to perform the types of analyses of interest and well trained in the use of the codes.

Source term analysis characterizes the radiological release to the environment resulting from each severe accident sequence leading to containment failure or bypass. The characterization includes the time, elevation, and energy of the release and the amount, form, and size of the radioactive material that is released to the environment. The source term analysis is sufficient to determine whether a large early release or a large late release occurs. A large early release is one involving the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions such that there is a potential for early health effects. Such accidents generally include unscrubbed releases associated with early containment failure at or shortly after vessel breach, containment bypass events, and loss of containment isolation. With large late release, unmitigated release from containment occurs in a timeframe that allows effective evacuation of the close-in population, making early health effects unlikely.

Quantification integrates the accident progression models and source term evaluation to provide estimates of the frequency of radionuclide releases that could be expected following the identified core damage accidents. This quantitative evaluation reflects the different magnitudes and timing of radionuclide releases and specifically allows for identification of LERF or LRF, as applicable.

Table 3 provides a summary of the characteristics and attributes needed for the technical elements for a Level 2 PRA for internal events. The characteristics and attributes are provided for at-power conditions.

Table 3. Summary of Technical Characteristics and Attributes of a Level 2, Internal Events PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
Plant Damage State Analysis	• Identification of the attributes of the core damage scenarios that influence severe accident progression, containment performance, and any subsequent radionuclide releases • Grouping of core damage scenarios with similar attributes into plant damage states • Carryover of relevant information from Level 1 to Level 2

Table 3. Summary of Technical Characteristics and Attributes of a Level 2, Internal Events PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
Severe Accident Progression Analysis	• Use of appropriate codes by qualified trained users with an understanding of the code limitations and the means for addressing the limitations • Assessment of the credible severe accident phenomena via a structured process • Assessment of containment system performance, including linkage with failure modes on noncontainment systems • Establishment of the capacity of the containment to withstand severe accident environments • Assessment of accident progression timing, including timing of loss of containment failure integrity
Source Term Analysis	• Assessment of radionuclide releases, including appreciation of timing, location, amount, and form of release • Grouping of radionuclide releases into smaller subsets of representative source terms, with emphasis on large early release and large late release
Quantification	• Estimation of the frequency of different containment failure modes and resulting radionuclide source terms

C.1.2.3 Technical Elements for an Internal Flood, At-Power Probabilistic Risk Assessment

This section identifies the technical elements for an internal flood, at-power PRA. The objective for each technical element is briefly described and the characteristics and attributes needed to achieve the objective are provided in Table 4. The technical elements for an internal flood, at-power PRA are:

- flood area partitioning,
- flood source analysis,
- flood scenario analysis, and
- flood scenario delineation and quantification.

PRA models of internal floods are based on an existing up-to-date internal events, at-power PRA model, which is modified to include the impact of the identified flood scenarios in terms of causing initiating events, and failing equipment used to respond to initiating events. The quantification task specific to internal floods is similar in nature to that for the internal events. Because of its dependence on the internal events model, the internal flood PRA incorporates the elements of regulatory positions C.1.2.1 and C.1.2.2, as necessary.

Flood area partitioning divides the plant into flood areas that are used as the basis for the flood analysis. Flood areas are defined on the basis of physical barriers, mitigation features, and propagation pathways.

Flood source analysis identifies the flood sources in each flood area that are attributable to equipment (e.g., piping, valves, pumps) and other sources internal to the plant (e.g., tanks) along with the affected SSCs. Flood mechanisms examined include failure modes of components, human-induced mechanisms, and other water-releasing events. Flood types (e.g., leak, rupture, spray) and flood sizes are determined. Plant walkdowns are performed to verify the accuracy of the information. It is recognized that at the design and initial licensing stages, plant walkdowns are not possible.

Flood scenario analysis identifies the potential flood scenarios for each flood source by identifying flood propagation paths of water from the flood source to its accumulation point (e.g., pipe and cable penetrations, doors, stairwells, failure of doors or walls). Plant design features or operator actions that have the ability to terminate the flood are identified. The susceptibility of each SSC in a flood area to flood-induced mechanisms is examined (e.g., submergence, spray, pipe whip, and jet impingement). Flood scenarios are developed by examining the potential for propagation and giving credit for flood mitigation. Flood scenarios can be eliminated on the basis of screening criteria. The screening criteria used are well-defined and justified.

Flood scenario delineation and quantification provide an estimation of the CDF and LERF (or LRF, as applicable) of the plant that includes internal floods. The frequency of flood-induced initiating events that represent the design, operation, and experience of the plant are quantified. The Level 1 internal events PRA is modified and the internal flood accident sequences are quantified to (1) modify accident sequence models to address flood phenomena, (2) perform necessary calculations to determine success criteria for flood mitigation, (3) perform parameter estimation analysis to include flood as a failure mode, (4) perform human reliability analysis to account for performance shaping factors that are attributable to flooding, and (5) quantify internal flood accident sequence CDF and LERF (or LRF, as applicable).

Table 4. Summary of Technical Characteristics and Attributes of an Internal Flood PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
Flood Area Partitioning	• Flood areas defined based on plant features that can restrict flood • Verification of area definitions through plant walkdowns
Flood Source Analysis	• Sufficiently detailed identification and characterization of the following: – SSCs located within each area – flood sources and flood mechanisms – type of water release and capacity • Use of well-defined and justified screening criteria for the elimination of flood sources and areas • Verification of the information through plant walkdowns for as-built plants
Flood Scenario Analysis	• Identification and evaluation of the following: – flood propagation paths – flood mitigating plant design features (e.g., drains and sumps) and operator actions – the susceptibility of SSCs in each flood area to the different types of floods • Use of well-defined and justified screening criteria for the elimination of flood scenarios
Flood Scenario Delineation and Quantification	• Identification and grouping of flood-induced initiating events on the basis of a structured and systematic process • Estimation of flood initiating event frequencies • Modification of the Level 1 internal events PRA to account for flooding effects, including uncertainties • Estimation of CDF and LERF (or LRF, as applicable) for chosen flood sequences • Use of well-defined and justified screening criteria for the elimination of flood scenarios

C.1.2.4 Technical Elements for an Internal Fire, At-Power Probabilistic Risk Assessment

This section identifies the technical elements for an internal fire, at-power PRA. The objective for each technical element is briefly described and the characteristics and attributes needed to achieve the objective are provided in Table 5. The technical elements for an internal fire, at-power PRA are the following:

- plant boundary definition and partitioning,
- equipment selection,
- cable selection,
- qualitative screening,
- fire PRA plant response model,
- fire scenario selection and analysis,
- fire ignition frequencies,
- quantitative screening,
- circuit failure,
- fire risk quantification, and
- fire/seismic interactions.

PRA models of internal fire are based on an existing up-to-date internal events, at-power PRA model, which is modified to reflect fire-induced failure of equipment contributing to or causing initiating events, fire-induced failure of equipment used to respond to initiating events, and the impact of fire on operator actions. Because of its dependence on the internal events model, the internal fire PRA incorporates the elements of regulatory positions C.1.2.1 and C.1.2.2 of this guide as necessary.

Plant boundary definition and partitioning establish the overall boundaries of the fire PRA and divides the area within that boundary into smaller regions (i.e., physical analysis units), commonly known as fire areas or compartments. The entire fire PRA is generally organized according to these physical analysis units.

Equipment selection identifies the equipment to be included in the fire PRA model. This equipment is selected from the equipment included in the internal events PRA and in the plant's fire protection program and analysis (i.e., the postfire safe-shutdown analysis) that, if failed by a fire, could produce a plant initiator or affect the plant response. Fire-induced spurious actuations are of particular interest. The selected equipment is mapped to the physical analysis units.

Cable selection identifies those cables associated with the equipment identified in the equipment selection technical element. The selected cables are mapped to the physical analysis units.

Qualitative screening is an optional element that may be used to eliminate certain physical analysis units defined in the plant boundary definition and partitioning element that can be shown to be unimportant to fire risk. General, qualitative criteria are typically applied. Those physical analysis units screened out in this technical element play no role in the more detailed quantitative assessment.

Fire PRA plant response models include a logic model that represents the plant response following a fire. This model is based upon the internal events PRA model, which is modified to account for fire effects. These modifications include SSC failures that specifically result from fire and consider fire-specific procedures. The latter are processed through the human reliability analysis technical element.

Fire scenario selection and analysis defines and analyzes fire event scenarios that capture the plant fire risk associated with each physical analysis unit. Fire scenarios are defined in terms of ignition sources, fire growth and propagation, fire detection, fire suppression, and cables and equipment ("targets") damaged by the fire. Main control room fire scenarios, including control room abandonment, are analyzed explicitly. Multicompartment fire propagation scenarios, including scenarios from all screened physical analysis units, are also assessed.

Fire ignition frequencies are estimated for the ignition sources postulated for the fire scenarios. Ignition sources consist of in situ sources, such as electrical cabinets or batteries, and other sources such as transient fires. U.S. nuclear power industry fire event frequencies, possibly augmented with plant-specific experience, are used where available to establish the fire ignition frequencies. Other sources are generally used only for cases when the U.S. nuclear power industry does not provide the representative frequency.

Quantitative screening involves eliminating physical analysis units from further quantitative analysis based on their quantitative contribution to fire risk. Quantitative screening criteria are established in terms of fire-induced CDF and LERF (or LRF, as applicable). This element is not required, although it is used in most applications. Note that, unlike the physical analysis units screened during qualitative screening, the CDF and LERF (or LRF, as applicable) contributions of each of these quantitatively screened units are retained and reported as a part of the total plant fire risk in the fire risk quantification element. All physical analysis units are reconsidered as a part of the multicompartment fire scenario analysis, regardless of the quantitative screening results.

Circuit failure analysis treats the impact of fire-induced circuit failures upon the plant response. In particular, spurious actuations from hot shorts (inter- and intra-cable) are analyzed. The conditional probability of the particular circuit failure is identified and assigned.

Post-fire human reliability analysis is conducted to identify operator actions and related HFEs, both within and outside the main control room, for inclusion in the plant response model. This element also includes quantification of HEPs for the modeled actions. Modeled operator actions include those introduced into the plant response model resulting strictly from fire-related emergency procedures and those actions retained from the internal events PRA. The latter HFEs are modified to account for fire effects.

Fire risk quantification calculates the fire-induced CDF and LERF (or LRF, as applicable) contributions to plant risk and identifies significant contributors to each. In this element, the plant response model is quantified for the set of fire scenarios to produce conditional core damage probability and conditional large early release probability (CLERP) or conditional large release probability (CLRP) values, as applicable. The conditional core damage probability and CLERP (or CLRP, as applicable) values are mathematically combined with the corresponding fire ignition frequencies and the conditional probabilities of fire damage for the appropriate fire scenario to yield fire-induced CDF and LERF (or LRF, as applicable).

Seismic/fire interactions are considered in a qualitative review of the plant fire risk caused by a potential earthquake. This element seeks to ensure that such seismic/fire interactions have been considered and their impacts assessed.

Table 5. Summary of Technical Characteristics and Attributes of an Internal Fire PRA for the At-Power Operating Mode

Element	**Technical Characteristics and Attributes**
Plant Boundary Definition and Partitioning	• The global analysis boundary captures all plant locations relevant to the fire PRA. • Physical analysis units are identified by credited partitioning elements that are capable of substantially confining fire damage behaviors.
Equipment Selection	• Equipment is selected for inclusion in the plant response model that leads to a fire-induced plant initiator, or that is needed to respond to such an initiator (including equipment subject to fire-induced spurious actuation that affects the plant response). • The number of spurious actuations to be addressed increases according to the significance of the consequence (e.g., interfacing systems LOCA). • Instrumentation and support equipment are included.
Cable Selection	• Cables that are required to support the operation of fire PRA equipment (defined in the equipment selection element) are identified and located.
Qualitative Screening (Optional Element)	• Screened-out physical analysis units represent negligible contributions to risk and are considered no further.
Fire PRA Plant Response Model	• Based upon the internal events PRA, the logic model is adjusted to add new fire-induced initiating events and modified or new accident sequences, operator actions, and accident progressions (in particular those from spurious actuations). • Inapplicable aspects of the internal events PRA model are bypassed.
Fire Scenario Selection and Analysis	• Fire scenarios are defined in terms of ignition sources, fire growth and propagation, fire detection, fire suppression, and cables and equipment (“targets”) damaged by fire. • The effectiveness of various fire protection features and systems is assessed (e.g., fixed suppression systems). • Appropriate fire modeling tools are applied. • The technical basis is established for statistical and empirical models in the context of the fire scenarios (e.g., fire brigade response). • Scenarios involving the fire-induced failure of structural steel are identified and assessed (at least qualitatively).
Fire Ignition Frequencies	• Frequencies are established for ignition sources and consequently for physical analysis units. • Transient fires should be postulated for all physical analysis units regardless of administrative controls. • Appropriate justification should be provided to use nonnuclear experience to determine fire ignition frequency.
Quantitative Screening	• Physical analysis units that are screened out from more refined quantitative analysis are retained to establish CDF and LERF (or LRF, as applicable). • Typically, those fire PRA contributions to CDF and LERF (or LRF, as applicable) that are established in the quantitative screening phase are conservatively characterized.

Table 5. Summary of Technical Characteristics and Attributes of an Internal Fire PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
Circuit Failure Analysis	• The conditional probability of occurrence of various circuit failure modes given cable damage from a fire is based upon cable and circuit features.
Post-fire Human Reliability Analysis	• Operator actions and related post-initiator HFEs, conducted both within and outside of the main control room, are addressed. • The effects of fire-specific procedures are identified and incorporated into the plant response model. • Plausible and feasible recovery actions, assessed for the effects of fire, are identified and quantified. • Undesired operator actions resulting from spurious indications are addressed. • Operator actions from the internal events PRA that are retained in the fire PRA are assessed for fire effects.
Fire Risk Quantification	• For each fire scenario, the fire risk results are quantified by combining the fire ignition frequency, the probability of fire damage, and the conditional core damage probability (and CLERP (or CLRP, as applicable)) from the fire PRA plant response model. • Total fire-induced CDF and LERF (or LRF, as applicable) are calculated for the plant and significant contributors identified. • The contribution of quantitatively screened scenarios (from the quantitative screening element) is added to yield the total risk values.
Seismic/Fire Interactions	• Potential interactions resulting from an earthquake and a resulting fire that might contribute to plant risk are reviewed qualitatively. • Qualitative assessment verifies that such interactions have been considered and that steps are taken to ensure that the potential risk contributions are mitigated.

C.1.2.5 Technical Elements for a Seismic, At-Power Probabilistic Risk Assessment

This section identifies the technical elements for a seismic, at-power PRA. The objective for each technical element is briefly described and the characteristics and attributes needed to achieve the objective are provided in Table 6. It is assumed that the seismic PRA is based on modifications made to an existing up-to-date internal events, at-power PRA. The technical elements for a seismic, at-power PRA are the following:

- seismic hazard analysis,
- seismic fragility analysis, and
- seismic plant response analysis.

Earthquakes can cause different initiating events than those considered in an internal events PRA and can cause simultaneous failures of multiple redundant components, an important common-cause effect that is included in a probabilistic seismic analysis. All possible levels of earthquakes along with their frequencies of occurrence and consequential damage to plant systems and components are considered in a probabilistic seismic analysis. Because of its dependence on the internal events model, the seismic PRA incorporates the elements of regulatory positions C.1.2.1 and C.1.2.2, as necessary.

Seismic hazard analysis is used to express the seismic hazard in terms of the frequency of exceedance for selected ground motion parameters during a specified time interval using a site-specific probabilistic hazard analysis that incorporates the available recent site-specific information and uses up-to-date databases. The analysis involves the identification of earthquake sources, the evaluation of the regional earthquake history, and an estimate of the intensity of the earthquake-induced ground motion at the site. At most sites, the objective is to estimate the probability or frequency of exceeding different levels of vibratory ground motion. However, in some cases, other seismic hazards are included, such as fault displacement, soil liquefaction, soil settlement, and earthquake-induced external flood. For all the various hazards, the objective is to estimate the probability or frequency of the hazard as a function of its intensity. The complexity of the hazard analysis depends on the complexity of the seismic situation at the site, as well as the ultimate intended use of the seismic PRA. Where no prior study exists, the site-specific probabilistic seismic hazard should be generated. However, in many cases, an existing study can be used to develop a site-specific probabilistic seismic hazard. In a probabilistic seismic hazard analysis, an essential part of the methodology is the consideration of both aleatory and epistemic uncertainties, which typically results in generating a set of hazard curves defined at specified fractile (confidence) levels and a mean hazard curve.

Seismic fragility analysis estimates the conditional probability of SSC failures at a given value of a seismic motion parameter such as peak ground acceleration, peak spectral acceleration, and floor spectral acceleration. Seismic fragilities used in a seismic PRA are realistic and plant-specific based on actual current conditions of the SSCs in the plant, as confirmed through a detailed walkdown of the plant. The fragilities of all the systems modeled in the accident sequences are included.

Seismic plant response analysis calculates the frequencies of severe core damage and radioactive release to the environment by combining the plant logic model with component fragilities and seismic hazard estimates. The analysis is usually carried out by using the internal events PRA model as the foundation and adding basic events for seismic-induced failures to the internal events PRA model. Some portions of the internal events PRA model that do not apply or that can be screened out based on the impact on the base seismic PRA should be eliminated. For example, recovery of offsite power is highly unlikely after a large earthquake, and therefore portions of the internal events model related to offsite power recovery can often be eliminated. Further screening out low-probability, nonseismic failures and human-error events may also be possible, although significant nonseismic failures and human errors must be included. Therefore, the seismic PRA model is usually adapted from the internal events, at-power PRA model to incorporate unique seismic-related aspects that are different from the at-power, internal events PRA model. In some cases, instead of starting with the internal events model and adapting it, a special seismic model is created from scratch. In this case, it is especially important to check for consistency with the internal events model regarding plant response and the cause-effect relationships of the failures. In any case, the seismic PRA model includes all significant seismic causes, initiating events, and seismic-induced SSC failures, as well as significant nonseismic failures and human errors. The model reflects the as-built and as-operated plant.

In meeting the technical characteristics and attributes for the seismic portion of an external hazard PRA, a seismic margins method is not an acceptable approach to the NRC staff because a seismic margins method does not define and quantify seismically-induced accident sequences. A seismic PRA developed by COL holders such that it meets the requirements of (1) ASME/ANS RA-Sa-2009 Part 5, as endorsed in Appendix A to this RG, or (2) ASME/ANS RA-S Case 1, as endorsed in Appendix B to this RG is an acceptable way to meet the seismic hazard portion of the fuel-load PRA requirements per 10 CFR 50.71(h)(1).

Table 6. Summary of Technical Characteristics and Attributes of a Seismic PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
Probabilistic Seismic Analysis	• Seismic hazard analysis – establishes the frequency of earthquakes at the site – uses site-specific data – examines all credible sources of damaging earthquakes – includes current information – based on comprehensive data, including o geological, seismological, and geophysical data o local site topography o historical information – reflects the composite distribution of the informed technical community. – level of analysis depends on application and site complexity • Aleatory and epistemic uncertainties in the hazard analysis (in characterizing the seismic sources and the ground motion propagation) – properly accounted for – fully propagated – allow estimates of o fractile hazard curves, o median and mean hazard curves, o uniform hazard response spectra • Spectral shape used in the seismic PRA – based on a site-specific evaluation – broad-band, smooth spectral shapes for lower-seismicity sites are acceptable if shown to be appropriate for the site – uniform hazard response spectra are acceptable if it reflects the site-specific shape • Assess whether for the specific application, other seismic hazards should be included in the seismic PRA, such as – fault displacement – landslide, – soil liquefaction – soil settlement
Seismic Fragility Analysis	• Seismic fragility estimate – plant-specific – realistic – includes all systems that participate in accident sequences included in the seismic-PRA systems model – basis for screening of high-capacity components is fully described • Seismic fragility evaluation performed for critical SSCs based on – review of plant design documents – earthquake experience data – fragility test data – generic qualification test data (use is justified) – analytical approaches using plant- and location-specific seismic demand information – walkdowns • walkdowns focus on as-built and as-operated plant configuration, including – anchorage

Table 6. Summary of Technical Characteristics and Attributes of a Seismic PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
	– lateral seismic support – potential systems interactions
Seismic Plant Response Analysis	• The seismic PRA models include – seismic-caused initiating events – seismic-induced SSC failures – nonseismic-induced unavailabilities, – other significant failures (including human errors) that can lead to CDF or LERF (or LRF, as applicable) • The seismic PRA models – adapted to incorporate seismic-analysis aspects that are different from corresponding aspects found in the at-power, internal events PRA model – reflects the as-built and as-operated plant being analyzed • Quantification of CDF and LERF (or LRF, as applicable) integrates – the seismic hazard – the seismic fragilities – the systems analysis

C.1.2.6 Technical Elements for a Screening and Conservative Analysis of Hazards for an At-Power Probabilistic Risk Assessment

Screening methods can often be employed to show that the contribution of a hazard to CDF, LERF (or LRF, as applicable), or both is not significant. Criteria that have been recognized for screening out are the following: A hazard can be screened out either (1) if it meets the criteria in the 1975 or later revision of NUREG-0800, (2) if it can be shown using a demonstrably conservative analysis that the mean value of the design-basis hazard used in the plant design is less than $1x10^{-5}$ per year and that the conditional core damage probability is less than $1x10^{-1}$, given the occurrence of the design-basis-hazard event, or (3) if it can be shown using a demonstrably conservative analysis that the CDF is less than $1x10^{-6}$ per year. It is recognized that for those new reactor designs with substantially lower risk profiles (e.g., internal events CDF below $1x10^{-6}$ per year), the quantitative screening value should be adjusted according to the relative baseline risk value.

Screening and conservative analysis is usually the first task an analyst performs when developing a base PRA. Natural and human-caused hazards that apply to the site under consideration are first identified. Table D-1 in Appendix D provides a list of hazards that should be addressed in the base PRA. Many of the hazards in Table D-1 may be eliminated from a detailed PRA if they can be screened out based on the screening criteria defined above. A preliminary screening, using a defined set of screening criteria, is used to eliminate from further consideration risk contributors matching the criteria. Further screening can be performed by using a bounding or demonstrably conservative analysis with defined quantitative screening criteria to demonstrate that the risk from some external events is sufficiently low to eliminate them from additional consideration. Walkdowns of the plant site and plant buildings are used to confirm the assumptions used for the screening basis. For some risk-informed applications, the validity of the assumptions used to screen out a hazard from the base PRA may need to be examined and confirmed by the staff, based on related application-specific guidance.

Table 7 summarizes the characteristics and attributes needed for the technical elements for screening and conservative analysis.

Table 7. Summary of Technical Characteristics and Attributes of Screening and Conservative Analysis of Hazards

Element	Technical Characteristics and Attributes
Screening and Conservative Analysis	• All potential hazards that can affect the site identified • Preliminary screening performed using a defined set of criteria • Bounding or conservative analysis performed using defined quantitative screening criteria • Basis for screening confirmed with a walkdown

C.1.2.7 Technical Elements for a High-Wind, At-Power Probabilistic Risk Assessment

This section identifies the technical elements for a high-wind, at-power PRA. The objective for each technical element is briefly described, and the characteristics and attributes needed to achieve the objective are provided in Table 8. The technical elements for a high-wind, at-power PRA are the following:

- high-wind hazards analysis,
- high-wind fragility analysis, and
- high-wind plant response analysis.

Screening methods can often be used to show that the contribution of high winds to CDF, LERF (or LRF, as applicable), or both is insignificant. The considerations in this section apply to those high-wind phenomena that have not been screened out. The types of high-wind events that should be considered in the analysis are site dependent. These can include tornados and their effects, cyclones, hurricanes, and typhoons, as well as thunderstorms, squall lines, and other weather fronts. It is assumed that the high-wind PRA is based on modifications made to an existing up-to-date internal events, at-power PRA. The technical elements for a high-wind PRA are similar to those for a seismic PRA. Because of its dependence on the internal events model, the high-wind PRA incorporates the elements of regulatory positions C.1.2.1 and C.1.2.2, as necessary.

High-wind hazard analysis estimates the frequency of high wind at the site using a site-specific probabilistic wind hazard analysis that incorporates the available recent regional and site-specific information and uses up-to-date databases. Uncertainties in the models and parameter values are properly accounted for and fully propagated to allow the derivation of a mean hazard curve from the family of hazard curves obtained.

High-wind fragility analysis is an evaluation that is performed to estimate plant-specific, realistic wind fragilities for those SSCs (or their combination) whose failure contributes to core damage or large early release (or large release).

High-wind plant response analysis uses a high-wind PRA systems model that includes all significant high-wind-caused initiating events and other failures that can lead to core damage or large early release (or large release). The model is adapted from the internal events, at-power PRA model to incorporate unique high-wind analysis aspects that are different from the at-power, internal events PRA model.

Table 8. Summary of Technical Characteristics and Attributes of a High Wind PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
High-Wind Hazard Analysis	• Probabilistic wind hazard analysis – results in frequency of high wind at the site – based on site-specific data – reflects recent information • Uncertainties in the models and parameter values – properly accounted for – fully propagated – allow estimate of mean hazard curve
High-Wind Fragility Analysis	• Wind fragility estimate – plant-specific – realistic – all SSCs whose failure contributes to core damage or large early release (or large release) included – walkdowns focus on as-built and as-operated plant configuration
High-Wind Plant Response Analysis	• Wind PRA model – includes all significant wind-caused initiating events – includes other significant failures (both those that are wind-caused and those that are random failures) that can lead to CDF or LERF (or LRF, as applicable) – adapted from the internal events, at-power PRA model – incorporates unique wind-analysis aspects that are different from the at-power, internal events PRA model

C.1.2.8 Technical Elements for an External Flood, At-Power Probabilistic Risk Assessment

This section identifies the technical elements for an external flood, at-power PRA. The objective for each technical element is briefly described, and the characteristics and attributes needed to achieve the objective are provided in Table 9. The technical elements for an external flood, at-power PRA are the following:

- external flood hazard analysis,
- external flood fragility analysis, and
- external flood plant response analysis.

Screening methods can often be employed to show that the contribution of some external flood events to CDF, LERF (or LRF, as applicable), or both is insignificant. The considerations in this section apply to those flood phenomena that have not been screened out. The analysis of how the flood pathways and water levels cause the failure of SSCs following ingress into the plant structures is similar to the analysis in the internal flood PRA. The technical elements for an external flood PRA are similar to those for an internal flood PRA and a seismic PRA. The types of external flood phenomena that should be considered in the analysis are dependent on the site. Both natural phenomena, such as river or lake flooding, ocean flooding from high tides or storm surges, unusually high precipitation, tsunamis, and seiches, as well as human-caused events such as failures of dams, levees, and dikes, should be considered. It is assumed that the external flood PRA is based on modifications made to an existing, up-to-date internal events, at-power PRA. Because of its dependence on the internal events model, the external flood PRA incorporates the elements of regulatory positions C.1.2.1 and C.1.2.2, as necessary.

External flood hazard analysis estimates the frequency of external flood at the site using a site-specific probabilistic hazard analysis that incorporates the available recent site-specific information and uses up-to-date databases. Uncertainties in the models and parameter values are properly accounted for and fully propagated to allow the derivation of a mean hazard curve from the family of hazard curves obtained.

External flood fragility analysis is an evaluation that is performed to estimate plant-specific, realistic flood fragilities for those SSCs (or their combination) whose failure contributes to core damage or large early release (or large release).

External flood plant response analysis uses an external flood PRA model that includes all significant flood-caused initiating events and other failures that can lead to core damage or large early release (or large release). The model is adapted from the internal events, at-power PRA model to incorporate unique flood-analysis aspects that are different from the at-power, internal events PRA model.

Table 9. Summary of Technical Characteristics and Attributes of an External Flood PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
External Flood Hazard Analysis	• Probabilistic flood hazard analysis - results in frequency of external flood at the site - based on site-specific data - reflects recent information • Uncertainties in the models and parameter values - properly accounted for - fully propagated - allow estimate of mean hazard curve
External Flood Fragility Analysis	• Flood fragility estimate - plant-specific - realistic - all SSCs whose failure contributes to core damage or large early release (or large release) included - walkdowns focus on as-built and as-operated plant configuration
External Flood Plant Response Analysis	• External flood PRA model - includes all significant flood-caused initiating events - includes other significant failures (both those that are caused by the flood and those that are random failures) that can lead to CDF or LERF (or LRF, as applicable) - adapted from the internal events, at-power PRA model - incorporates unique flood-analysis aspects that are different from the at-power, internal events PRA model

C.1.2.9 Technical Elements for an Other Hazards, At-Power Probabilistic Risk Assessment

This section identifies the technical elements for an other hazards, at-power PRA. As discussed in regulatory position C.1.1, other hazards are those hazards that are not categorized under the internal events, internal flood, internal fire, seismic, high-wind, or external flood hazards groups and cannot be screened out by a screening analysis, as described in regulatory position C.1.2.6. The objective for each technical element is briefly described, and the characteristics and attributes needed to achieve the objective are provided in Table 10. The technical elements for an other hazards, at-power PRA are the following:

- other hazard analysis,
- other hazard fragility analysis, and
- other hazard plant response analysis.

Screening methods can often be employed to show that the contribution of a hazard to CDF, LERF (or LRF, as applicable), or both is insignificant. The considerations in this section apply to those hazards identified in Table D-1 of Appendix D that are not screened out based on a screening and conservative analysis. It should be noted that because of the limited collective experience of the analysis community in the area of PRA for other hazards, an extensive peer review is particularly important for such a PRA. PRA models of other hazards are based on an existing up-to-date internal events, at-power PRA that is modified to include the impact of the hazard under consideration. Because of its dependence on the internal events model, the other hazard analysis incorporates the elements of regulatory positions C.1.2.1 and C.1.2.2, as necessary.

Other hazard analysis establishes the frequency of occurrence of different intensities of the hazard being analyzed and uses a site-specific probabilistic evaluation that is based on recent available data and site-specific information. Historical data or a phenomenological model, or a mixture of the two is used in the analysis.

Other hazard fragility analysis is an evaluation that is performed to estimate the fragility or vulnerability of an SSC (or their combination) whose failure contributes to core damage or large early release (or large release). The fragility analysis uses plant-specific information and an accepted engineering method for evaluating failures.

Other hazard plant response analysis uses a model that includes all important initiating events and other important failures caused by the effects of the hazard that can lead to core damage or large early release (or large release). The model is adapted from the internal events, at-power PRA model to incorporate unique aspects related to the hazard analyzed that are different from the at-power, internal events PRA model.

Table 10. Summary of Technical Characteristics and Attributes of an Other Hazard PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
Other Hazard Analysis	• Probabilistic hazard analysis – results in the hazard’s frequency of occurrence at the site – based on site-specific data – reflects recent information – uses historical data or a phenomenological model, or a mixture of the two
Other Hazard Fragility Analysis	• Fragility estimate – plant-specific – SSC-specific information – uses accepted engineering methods – walkdowns, if applicable, focus on as-built and as-operated plant configuration
Other Hazard Plant Response Analysis	• Hazard model – includes all important initiating events related to the hazard analyzed – includes other significant failures that can lead to CDF or LERF (or LRF, as applicable) – adapted from the internal events, at-power PRA model

Table 10. Summary of Technical Characteristics and Attributes of an Other Hazard PRA for the At-Power Operating Mode

Element	Technical Characteristics and Attributes
	– incorporates unique aspects related to the hazard analyzed that are different from the at-power, internal events PRA model

C.1.2.10 Technical Elements for Level 1 and Level 2 Low-Power and Shutdown Probabilistic Risk Assessments

The objectives of the technical elements for Level 1 or Level 2 LPSD PRAs are similar to the objectives of the technical elements for a Level 1, internal events, at-power PRA and a Level 2, internal events, at-power PRA, as described in regulatory positions C.1.2.1 and C.1.2.2, respectively. Table 11 provides the technical characteristics and attributes needed for accomplishing the objectives of the technical elements, with respect to Level 1 and Level 2 LPSD PRAs.

Table 11. Summary of Technical Characteristics and Attributes of Level 1 and Level 2, Internal Events PRAs for the Low-Power and Shutdown Operating Modes

Element	Technical Characteristic and Attributes
Level 1 Internal Events	
Plant Operating States	• The Level 1 PRA involves identification and characterization of a set of plant operational states during LPSD operations that are representative of all the plant states not covered in the full-power PRA. • The LPSD evolution is divided into POSs based on the unique impact on plant response to facilitate the practicality and efficiency of the PRA. • Each LPSD POS required to be considered for the specific application is identified and characterized as to all important conditions affecting the delineation and evaluation of core damage and large early release (or large release). • The conditions include decay heat level, reactor coolant system configuration, reactor level, pressure and temperature, containment configuration, and the assumed representative plant system configurations within the POS. • LPSD POSs that are subsumed into each other are shown to be represented by the characteristics of the subsuming group. • The duration and number of entries into each POS are determined. • The development, grouping, and quantification of the POSs are documented in a manner that facilitates PRA applications, upgrades, and peer review.
Initiating Event Analysis	• The initiating event analysis includes the same characteristics and attributes as for at-power, as well as the following: – examination of human-induced initiating events, for example, those resulting from maintenance activities, including different types of LOCAs (e.g., drain-down events as opposed to pipe breaks) – review of plant operational practices in grouping of events
Success Criteria Analysis	• The success criteria analysis includes the same characteristics and attributes as for at-power, as well as an analysis appropriate to the POS definition and characterization.
Accident Sequence Development Analysis	• The accident sequence development analysis includes the same characteristics and attributes as for at-power, as well as an accounting for changing plant conditions within a POS.

Table 11. Summary of Technical Characteristics and Attributes of Level 1 and Level 2, Internal Events PRAs for the Low-Power and Shutdown Operating Modes

Element	Technical Characteristic and Attributes
Systems Analysis	• The systems analysis includes the same characteristics and attributes as for at-power, as well as the identification of conditions varying from one POS to another for spatial and environmental hazards, systems actuation signals, and system inventories (e.g., air).
Parameter Estimation Analysis	• The parameter estimation analysis includes the same characteristics and attributes as for at-power, as well as the following: – performance of estimation on a POS-specific basis, when necessary – consideration of plant-specific data unique to POS (i.e., not applicable to at-power)
Human Reliability Analysis	• The human reliability analysis includes the same characteristics and attributes as for at-power, as well as the following: – differentiation between calibration errors that may impact equipment performance at-power versus low-power and shutdown POSs – increased emphasis on contributions to initiating events – performance of the analysis on a POS basis – identification of dependent HFEs, particularly between those resulting in initiating events and those associated with responses to the initiating events – justification for credit of operator actions credited for recovery in slowly developing scenarios (e.g., recovery times greater than 24 hours)
Quantification	• Quantification includes the same characteristics and attributes for at-power, as well as the estimation of CDF and LERF (or LRF, as applicable) for each POS
Level 2 Internal Events	
Plant Damage State Analysis	• The plant damage state analysis includes the same characteristics and attributes as for at-power
Severe Accident Progression Analysis	• The severe accident progression analysis includes the same characteristics and attributes as for at-power, as well as the following: – estimation of containment capacity based on the capacity of temporary closure, although for some POSs, containment may be open or have a reduced pressure capability – assessment of the feasibility of the ability of operators to close containment before (1) adverse environmental conditions (e.g., temperature, radiation, humidity, noise) prevent closure or (2) core cooling is impacted
Quantification	• Quantification includes the same characteristics and attributes as for at-power.
Source Term Analysis	• The source term analysis includes the same characteristics and attributes as for at-power.
NOTE: (1) For LPSD conditions, the following characteristics and attributes are also needed: • verification of temporary alignments for the specific outage or average modeled outage for data collection • identification of existing flood barriers that may be impaired or disabled that could impact the flood zone • consideration of automatic responses that may differ from at-power conditions • identification of fire barriers that may be breached that could impact fire propagation between fire areas and fire zones	

C.1.2.11 Technical Elements for the Interpretation of Results (Including Uncertainty Analysis)

The results of the Level 1 PRA are examined to identify the contributors sorted by hazard group, initiating events (e.g., transients, LOCAs), or specific hazard plant damage states (e.g., fire scenarios, internal flood scenarios, seismic plant damage states), accident sequences, equipment failures, and human errors. Methods such as importance measure calculations (e.g., Fussell-Vesely Importance, risk achievement worth, risk reduction worth, and Birnbaum Importance) are used to identify the contributions of various events to the estimation of CDF for both individual sequences and the total CDF (i.e., both contributors to the total CDF, including the contribution from the different hazard groups and different operating modes (i.e., at-power and LPSD) and contributors to each contributing sequence, are identified).

The results of the Level 2 PRA are examined to identify the contributors (e.g., containment failure mode, physical phenomena) to the model estimation of LERF or LRF, as applicable, for both individual sequences and the model as a whole, using such tools as importance measure calculations (e.g., Fussell-Vesely Importance, risk achievement worth, risk reduction worth, and Birnbaum Importance).

For many applications, it is necessary to combine the PRA results from different hazard groups (e.g., from internal events, internal fire, and seismic events). For this reason, an important aspect in interpreting the PRA results is understanding both the level of detail associated with the modeling of each of the hazard groups and the hazard group-specific model uncertainties. With respect to the level of detail, for example, the analysis of specific scope items such as internal fire, internal flood, or seismic events typically involves a successive screening approach, so that more detailed analysis can focus on the more significant contributions. The potential conservatism associated with the evaluation of the less significant contributors using this approach is assessed for each hazard group.

In addition, each of the hazard groups has unique sources of uncertainty that can influence the insights derived from the PRA model. These sources can be a result of uncertainties in the parameter values used to quantify the PRA or from the models used to reflect the phenomena associated with the severe accident progression delineated in the PRA. Uncertainty and sensitivity analyses identify and characterize sources of uncertainty as well as the potential sensitivities of the results to related assumptions and modeling approximations.

Assumptions made in response to sources of model uncertainty and any conservatisms introduced by the analysis approaches can bias the assessment of importance measures with respect to the combined risk assessment and the relative contributions of the hazard groups to the various risk metrics. Therefore, the sources of model uncertainty are identified and their impact on the results analyzed for each hazard group individually, so that, when it is necessary to combine the PRA results, the overall results can be characterized appropriately. The sensitivity of the model results to key assumptions and sources of uncertainty from modeling assumptions, model boundary conditions, and other assumptions is evaluated. Such evaluations should include sensitivity analyses to look at key assumptions both individually and in logical combinations. The combinations analyzed are chosen to account for interactions among the variables. NUREG-1855 provides guidance on the treatment of uncertainties associated with PRA as well as guidelines regarding defining, identifying, and characterizing different sources of uncertainty.

Table 12 summarizes the characteristics and attributes needed for the technical elements of interpretation of results.

Table 12. Summary of Technical Characteristics and Attributes for the Interpretation of Level 1 and Level 2 PRA Results

Element	Technical Characteristics and Attributes
Level 1 PRA	
Interpretation of Results	• Identification of the significant contributors to CDF (hazard groups, initiating events, specific hazard plant damage states, accident sequences, equipment failures and human errors) • Identification of sources of uncertainty and their potential impact on the PRA model • Understanding of the impact of the assumptions on the CDF and the identification of the accident sequence and the accident sequence contributors
Level 2 PRA	
Interpretation of Results	• Identification of the contributors to containment failure, resulting source terms, and LERF (or LRF, as applicable) • Identification of sources of uncertainty and their impact on the PRA model • Understanding of the impact of the assumptions on Level 2 results

C.1.2.12 Technical Elements for Probabilistic Risk Assessment Documentation

The documentation of the PRA model provides the necessary information so that the results can easily be reproduced and justified. The sources of information used in the PRA also should be referenced and retrievable. The methodology used to perform each aspect of the work is described either through documenting the actual process in the PRA documentation or through reference to existing methodology documents. Sources of uncertainty (both parameter and model) are identified and their impact on the results are assessed generally for each technical element. A source of model uncertainty is one that is related to an issue for which there is no consensus approach or model (e.g., choice of data source, success criteria, reactor coolant pressure seal LOCA model, human reliability model). A key source of uncertainty is where the choice of approach or model is known to have an impact on the risk profile (e.g., total CDF and total LERF (or total LRF, as applicable)), the set of initiating events and accident sequences that contribute most to CDF and LERF (or LRF, as applicable), such that it influences a decision being made using the PRA. Assumptions made in performing the analyses are identified and documented along with their justification to the extent that the context of the assumption is understood. The results (e.g., products and outcomes) from the various analyses are documented.

Table 13 summarizes the characteristics and attributes needed for documentation for each hazard group and associated technical elements.

Table 13. Summary of Technical Characteristics and Attributes for PRA Documentation

Element	Technical Characteristics and Attributes
Traceability and Justification	• The documentation is sufficient to facilitate independent peer reviews. • The documentation describes the interim results (sufficient to provide traceability and defensibility of the final results) and the final results, insights, and sources of uncertainties. • The walkdown process, where applicable, and results are fully described.

C.1.3 Level of Detail of a Base Probabilistic Risk Assessment

The level of detail of a base PRA is defined in terms of the resolution of the modeling used to represent the behavior and operations of the plant. A minimal level of detail is necessary to ensure that the impacts of designed-in dependencies (e.g., support system dependencies, functional dependencies, and dependencies on operator actions) are correctly captured. This minimal level of detail is implicit in the technical elements comprising the base PRA and their associated characteristics and attributes.

For each given technical element, the level of detail modeled in the base PRA may vary. The detail may vary from the degree to which (1) plant design and operation are modeled, (2) specific plant experience is incorporated into the model, and (3) realism is incorporated into the analyses that reflect the expected plant response. Regardless of the level of detail developed in the base PRA, the technical characteristics and attributes should be addressed. That is, each characteristic and attribute is always addressed, but the degree to which it is addressed may vary. In general, the level of detail for the base PRA should be consistent with current good practice.[4]

The level of detail needed in a PRA that supports a risk-informed decision is dependent on the application. The application may involve using the PRA during different plant “stages” (i.e., design, construction, and operation). Consequently, a PRA used to support a DC may not have the same level of detail as a PRA of a plant that has several years of operating experience. While it is recognized that the level of detail may vary depending on the application, each technical element and its attributes should be addressed.

C.1.4 Plant Representation and Probabilistic Risk Assessment Configuration Control

Plant representation is defined in terms of how closely the base PRA represents the plant as it is actually built and operated. In general, PRA results used to support an application must be derived from a base PRA model that represents the as-built and as-operated plant to the extent needed to support the application. Consequently, the base PRA is maintained and upgraded, where necessary, to ensure it represents the as-built and as-operated plant. However, for some applications, the plant may only be in the DC or COL stage of licensing or may be under construction, at which points the plant may not have been built or is not yet operational. At these licensing stages, the base PRA model is intended to reflect the as-designed, as-to-be-built, and as-to-be-operated plant.

A PRA application is a documented analysis based in part or in whole on a design-specific or plant-specific PRA that is used to assist in decisionmaking with regard to the design, licensing, procurement, construction, operation, or maintenance of a nuclear power plant. In the context of regulatory activities, a PRA application includes the use of PRA results to support decisions related to any regulated activity regardless of whether the NRC or the licensee or applicant makes the decision.

Therefore, a process for developing, maintaining, and upgrading an acceptable base PRA is established. This process involves identifying and using plant information to develop and modify the base PRA, including changes to the plant that necessitate changes to the base PRA. The licensee or applicant will consider the cumulative impact of those plant changes or PRA model improvements, as needed, on the results of the PRA and applications being performed between any periodic update of the PRA; changes that would impact risk-informed decisions are addressed in the context of the application or implemented prior to the application. The process is performed such that the plant information identified and used in the base PRA reflects the as-designed, as-built, and as-operated plant and is as realistic as

4 The glossary in this RG provides a definition of the term current good practice.

possible in assessing the risk. The information sources include the applicable design, operation, maintenance, and engineering characteristics of the plant.

For those SSCs and human actions used in the development of the base PRA, the following information is identified, integrated, and used in the base PRA:

- plant design information reflecting the normal and emergency configurations of the plant
- plant operational information with regard to plant procedures and practices
- plant test and maintenance procedures and practices
- engineering aspects of the plant design

Further, plant walkdowns are conducted to ensure that information sources being used actually reflect the plant's as-built and as-operated condition. In some cases, corroborating information obtained from the documented information sources for the plant and other information may only be gained by direct observations. It is recognized that at the design and initial licensing stages, plant walkdowns are not possible.

Table 14 describes the technical characteristics and attributes needed for the above types of information.

Table 14. Summary of Technical Characteristics and Attributes for Information Sources Used in the Base PRA Development

Type of Information	**Characteristics and Attributes** (see note)
Design	• The safety functions required to maintain the plant in a safe, stable state and prevent core or containment damage • Identification of those SSCs that are credited in the base PRA to perform the above functions • The functional relationships among the SSCs, including both functional and hardware dependencies • The normal and emergency configurations of the SSCs • The automatic and manual (human interface) aspects of equipment initiation, actuation, and operation, as well as isolation and termination • The SSC's capabilities (flows, pressures, actuation timing, environmental operating limits) • Spatial layout, sizing, and accessibility information related to the credited SSCs • Other design information needed to support the base PRA modeling of the plant
Operational	• That information needed to reflect the actual operating procedures and practices used at the plant, including when and how operators interface with plant equipment as well as how plant staff monitor equipment operation and status • That information needed to reflect the operating history of the plant as well as any events involving significant human interaction
Maintenance	• That information needed to reflect planned and typical unplanned tests and maintenance activities and their relationship to the status, timing, and duration of the availability of equipment • Historical information related to the maintenance practices and experience at the plant

Table 14. Summary of Technical Characteristics and Attributes for Information Sources Used in the Base PRA Development

Type of Information	Characteristics and Attributes (see note)
Engineering	• The design margins in the capabilities of the SSCs • Operating environmental limits of the equipment • Expected thermal hydraulic plant response to different states of equipment (such as for establishing success criteria) • Other engineering information needed to support the base PRA modeling of the plant
It is recognized that for reactors in the design or construction stage, the level of operational and maintenance information may vary.	

As a plant operates over time, its associated risk may change. This change may occur for the following reasons:

- Operating data may change the availability or reliability of the plant's SSCs.
- Plant design or operation may change.
- The base PRA model may change as a result of improved methods or techniques.

Therefore, to ensure the base PRA represents the risk of the current as-built and as-operated plant, the base PRA is maintained and upgraded over time. Table 15 provides the characteristics and attributes needed for an acceptable process for maintaining and upgrading the base PRA. In addition, Appendix C provides guidance on whether the change to the PRA model is to be considered PRA maintenance or a PRA upgrade. COL holders should meet all requirements in ASME/ANS RA-Sa-2009, as endorsed in Appendix A to this RG, when the PRA is updated as required by 10 CFR 50.71(h)(2) and 10 CFR 50.71(h)(3).

Table 15. Summary of Technical Characteristics and Attributes for a PRA Configuration Control Process

Characteristics and Attributes
• Monitor PRA inputs and collect new information. • Ensure cumulative impact of pending plant changes are considered. • Maintain configuration control of the computer codes used in the base PRA. • Identify when the base PRA model should be updated based on new information or new models, techniques, and tools. • Ensure peer review is performed on PRA upgrades.

C.2 National Consensus Standards and Industry Programs for Probabilistic Risk Assessment

One acceptable approach to demonstrate conformance with regulatory position C.1 is to use a national consensus PRA standard or standards, as endorsed by the NRC staff with exceptions and clarifications, that address the scope of the base PRA. ASME and ANS have issued the ASME/ANS Level 1/LERF PRA standard, ASME/ANS RA-Sa-2009, which provides process and technical requirements for an at-power, Level 1 and limited Level 2 LWR PRA for internal hazards (i.e., internal events, internal floods, and internal fire), and external hazards (i.e., seismic, high wind, external flood, and other external events). ASME and ANS have also issued ASME/ANS RA-S Case 1, which provides a proposed alternative set of requirements related to the requirements for seismic PRA in Part 5 of the ASME/ANS Level 1/LERF PRA standard. These national consensus PRA standards establish requirements for what an acceptable base PRA should include, including acceptable process requirements; however, these PRA standards do not address how to meet the requirements for an acceptable PRA. Because ASME and ANS joint national consensus PRA standards as well as PWROG-19027-NP, Revision 2, use the term "requirement," "require," and other similar mandatory language, the staff's endorsement, including staff exceptions and clarifications, mirrors this language. The use of this language in this RG is not intended to convey that compliance with this RG is mandatory, or, provides the only way to meet the statutory and regulatory requirements, or that these requirements would be applied to licensees absent their adoption and consistent with the requirements of 10 CFR 50.109, "Backfitting."

Regulatory position C.2.1 of this RG provides the staff position on the use of a national consensus PRA standard to meet regulatory position C.1. To demonstrate acceptability of the PRA, a peer review is needed to determine whether the intent of the requirements in the national consensus PRA standard is met, as endorsed by the NRC staff with exceptions and clarifications, so that it can be demonstrated that the base PRA model is in conformance with regulatory position C.1. Regulatory position C.2.2 of this RG provides the staff position on the use of PRA peer reviews to this effect, including staff endorsement with exceptions and clarifications, of related industry PRA peer review guidance. When the peer review accounts for regulatory position C.2.2 and the PRA is assessed against a national consensus PRA standard consistent with regulatory positions C.1 and C.2.1, including staff exceptions and clarifications, the PRA is considered to be acceptable by the NRC staff for supporting that risk-informed regulatory application.

C.2.1 National Consensus Probabilistic Risk Assessment Standards

In general, if a national consensus PRA standard is used to demonstrate conformance with regulatory position C.1, the national consensus PRA standard should be based on a predetermined set of principles and objectives. Table 16 provides an acceptable set of principles and objectives that were established and used by ASME and ANS in development of their PRA standards.

Table 16. Principles and Objectives of a National Consensus PRA Standard

1. The PRA standard provides well-defined criteria against which the strengths and weaknesses of the PRA may be judged so that decision-makers can determine the degree of reliance that can be placed on the PRA results of interest.

2. The standard is based on current good practices as reflected in publicly available documents. The need for the documentation to be publicly available follows from the fact that the standard may be used to support safety decisions.

3. To facilitate the use of the standard for a wide range of applications, categories can be defined to aid in determining the applicability of the PRA for various types of applications.

4. The standard thoroughly and completely defines what is technically required and, where appropriate, identifies one or more acceptable methods.

5. The standard requires a peer review process that identifies and assesses where the technical requirements of the standard are not met. The standard needs to ensure that the peer review process meets the following criteria:

- determines whether methods identified in the standard have been used appropriately
- determines that, when acceptable methods are not specified in the standard, or when alternative methods are used in lieu of those identified in the standard, the methods used are adequate to meet the requirements of the standard
- assesses the significance of the results and insights gained from the PRA of not meeting the technical requirements in the standard
- highlights assumptions that may significantly impact the results and provides an assessment of the reasonableness of the assumptions
- is flexible and accommodates alternative peer review approaches
- includes a peer review team that is composed of members who are knowledgeable in the technical elements of a PRA, are familiar with the plant design and operation, and are independent with no conflicts of interest *that may influence the outcome of the peer review* [this clause was not in the ASME definition]

6. The standard addresses the maintenance and update of the PRA to incorporate changes that can substantially impact the risk profile so that the PRA adequately represents the current *as-designed* [added], as-built and as-operated plant.

7. The standard is a living document. Consequently, it should not impede research. It is structured so that, when improvements in the state of knowledge occur, the standard can easily be updated.

Principle 3 recognizes that the technical requirements of a PRA can be, and generally are, performed to different "capabilities." In developing the various models in the PRA, the different capabilities are distinguished by three attributes, determined by the degree to which the following criteria are met:

1. the scope and level of detail that reflects the plant design, operation, and maintenance;

2. the use of plant-specific information versus generic information to represent the as-designed, as-built, and as-operated plant; and

3. the degree of realism that is incorporated into the base PRA model to reflect the expected response of the plant.

It is recognized that a PRA may not satisfy each technical requirement to the same degree (i.e., capability category as used in an ASME/ANS PRA standard); that is, the capability category achieved for the different technical requirements may vary. This variation can range from (1) the minimum needed to meet the characteristics and attributes for each technical element to (2) the minimum to meet current good practice (i.e., state-of-practice) for each technical element. Further, which capability category is needed to be met for each technical requirement is dependent on the specific application. In general, the staff anticipates that current good practice (i.e., Capability Category II of an ASME/ANS PRA standard) is the level of detail that is acceptable for the majority of applications. However, for some applications, Capability Category I may be acceptable for some requirements. There may be other situations where the state-of-practice is not acceptable for a specific application. For example, it is not state-of-practice to include pipe failures in a base PRA model; however, for changes to inservice inspection and pressure boundary repair and replacement requirements, inclusion of pipe failures is needed. Therefore, it cannot be assumed that the state-of-practice is acceptable for every risk-informed application.

The requirements in an ASME/ANS PRA standard are either process-related or are technical in nature. The process-related requirements address the process for application, development, maintenance and upgrade, and peer review. The technical requirements address the technical elements of the PRA and what is necessary to acceptably perform that element.

For process requirements, the intent is generally straightforward and the requirement is either met or not met. For the technical requirements, it is not always as straightforward. Many of the technical requirements in an ASME/ANS PRA standard are applied more than once in developing the PRA model. For example, the requirements for systems analysis in a Level 1, internal events, at-power PRA apply to all systems modeled, and certain data requirements apply to all parameters for which estimates are provided. If, among these systems or parameter estimates there are instances where a specific requirement has not been met, it is not necessarily indicative that this requirement has not been met. If the requirement has been met for the majority of the systems or parameter estimates, and the identified instances are understood to be isolated mistakes or oversights, the requirement would be considered to be met. If, however, there is a systematic failure to address the requirement (e.g., component boundaries have not been defined anywhere), then the requirement has not been met. In either case, the instances of noncompliance with the PRA standard requirements are to be (1) rectified or demonstrated not to be relevant to the application and (2) documented.

Further, the technical requirements may be defined at two different levels: (1) high-level requirements (HLRs) and (2) supporting requirements (SRs). HLRs are defined for each technical element and capture the objective of the technical element. HLRs are defined in general terms, should be met regardless of the capability category, and accommodate different approaches. SRs are defined for each HLR and are the minimal requirements needed to satisfy the HLR. Consequently, a determination of whether an HLR is met is based on whether the associated SRs are met. Whether every SR is needed for an HLR depends on the application and is determined by the application process requirements. All SRs related to NDMs should be evaluated during peer reviews of NDMs. If any SRs are determined to be inapplicable, justification for such a conclusion should be documented.

C.2.2 Industry Peer Review Program

A peer review of the PRA is performed to determine whether the requirements established in the national consensus PRA standard, as endorsed by the NRC with exceptions and clarifications, have been met. An acceptable peer review approach is one that is performed according to an established process and by qualified personnel, documents the results, and identifies both strengths and weaknesses of the PRA. The ASME/ANS Level 1/LERF PRA standard requires a peer review to be performed on the base PRA model, any PRA upgrades, and the use of any NDMs. The ASME/ANS Level 1/LERF PRA standard provides requirements for (1) the establishment of a peer review process and (2) PRA peer review team qualifications and documentation. A peer review methodology (i.e., process) is documented in the industry-developed peer review guidance documents.

As stated earlier, the peer review is to be performed against established national consensus PRA standards (e.g., the ASME/ANS Level 1/LERF PRA standard, as endorsed by the NRC with exceptions and clarifications). If different criteria are used, other than those in an established national consensus PRA standard, then it should be demonstrated how these different criteria are consistent with the established national consensus PRA standards endorsed by the NRC with exceptions and clarifications.

For example, a peer review may be performed on the base PRA model or on a PRA upgrade, which may involve use of an NDM, or in the form of an independent assessment reviewing the closure of facts and observations (F&Os) from a peer review. Closure of F&Os could enhance the efficiency of NRC reviews of risk-informed applications that use PRA models. F&Os that are not closed using an NRC-endorsed process should be evaluated by the licensee or applicant for their impact on a risk-informed application and addressed with documented justification with necessary changes made to the PRA prior to the use of PRA in the risk-informed application. The following sections provide guidance on each of these scenarios.

C.2.2.1 Peer Review of a Base Probabilistic Risk Assessment Model

The **peer review process** includes a documented procedure used to direct the peer review team's evaluation of the acceptability of a base PRA. The review process compares the base PRA against established criteria (e.g., technical requirements defined in a national consensus PRA standard that conform to the PRA characteristics and attributes such as those provided in regulatory position C.1.2). In addition to reviewing the methods used in the base PRA, the peer review determines whether the methods were applied correctly. The base PRA models are compared against the plant design and procedures to validate that they reflect the as-designed, as-to-be-built, or the as-built and as-operated plant. Independent walkdowns should be performed to confirm PRA inputs, especially for external hazard PRAs. Assumptions are reviewed to determine whether they are appropriate and to assess their impact on the base PRA results. The base PRA results should be checked for fidelity with the model structure and for consistency with the results from PRAs for similar plants based on the peer reviewer's knowledge. Finally, the peer review process examines the procedures or guidelines established for updating the base PRA to reflect changes in plant design, operation, or experience. The process also provides criteria ensuring that the peer review is current. That is, (1) the peer review addresses the modifications made to the base PRA since any previous peer reviews, and (2) the peer review addresses modifications made to the standard since any previous peer reviews.

The **team qualifications** determine the credibility and acceptability of the peer reviewers. To avoid any perception of a technical conflict of interest, it is best if the members of the peer review team be prohibited from peer reviewing any portion of the PRA on which they have performed or supervised work or on which a direct supervisor performed or supervised work. Additionally, each member of the peer review team should have technical expertise (e.g., experience in performing (not just reviewing) the work in the PRA technical element assigned for review) in the PRA elements they review, including experience in the specific methods used to develop technical elements of the base PRA. Knowledge of the key features specific to the plant design and operation is essential. Finally, each member of the peer review team should be knowledgeable about the peer review process, including the desired characteristics and attributes used to assess the acceptability of the base PRA.

Documentation provides the necessary information to ensure that the peer review process and the findings are traceable, and the bases of the findings are defensible. Descriptions of the qualifications of the peer review team members and the peer review process should be documented. The results of the peer review for each technical element and the base PRA update process are described, including the areas in which the base PRA does not meet or exceed the desired characteristics and attributes used in the review process. This includes an assessment of the importance of any identified deficiencies in the base PRA results and how these deficiencies were addressed and resolved.

Table 17 summarizes the characteristics and attributes needed for an acceptable PRA peer review process. These characteristics and attributes are applicable to the types of PRA peer review discussed in regulatory positions C.2.2.1 through C.2.2.3.

Table 17. Summary of the Characteristics and Attributes of a PRA Peer Review

Element	Characteristics and Attributes
Peer Review Process	• Uses documented process • Uses as a basis for review a set of desired PRA characteristics and attributes • Uses a minimum list of review topics to ensure coverage, consistency, and uniformity • Reviews PRA methods (NDMs, as defined in the Glossary of this RG, should be reviewed against the NDMs requirements referenced in regulatory position C.2.2.2.2) • Reviews application of methods • Reviews assumptions and assesses their validity and appropriateness • Determines whether the PRA represents as-built and as-operated plant or the as-to-be-built and as-to-be-operated plant in the case of a DC or COL application.[5] • Performs independent walkdowns of the plant to confirm PRA inputs when information about plant configuration, or other aspects of the plant (e.g., spatial aspects) modeled in the PRA, are important to the development of PRA inputs and the acceptability of the base PRA (e.g., as related to internal flood, internal fire, seismic, high-wind, or external flood PRA, or other PRAs of hazards that are dependent on spatial information) • Reviews results of each PRA technical element for reasonableness • Reviews the PRA maintenance and update process • Reviews PRA modification attributable to the use of different models, techniques, or tools • Reviews against modifications to the standard

5 The NRC recognizes there may be special circumstances where credit for planned modification(s) in the PRA model is appropriate and these applications are addressed by the staff on a case-by-case basis. Regarding the peer review of a base PRA model that credits planned modifications to an operating plant, licensees or applicants should provide details of the planned modifications to the peer review team in advance of the peer review. These details should include any spatial information associated with the planned modification that may impact, for example, internal flood, internal fire, seismic, high wind, or external flood PRA, or other PRAs of hazards that are dependent on spatial information. The resulting peer review report should clearly identify any planned modifications reviewed by the peer review team. Regulatory position C.4.2 provides guidance on submittal documentation for such cases.

Table 17. Summary of the Characteristics and Attributes of a PRA Peer Review

Element	Characteristics and Attributes
Team Qualifications	• Independent with no conflicts of interest (i.e., should be prohibited from peer reviewing any portion of the PRA they have performed or supervised work on) • Individuals are not assigned an area for review where their current immediate supervisor performed or supervised the actual technical analysis • Collectively represent expertise in all the technical elements of a base PRA, including integration • Expertise in the technical element assigned to review • Knowledge of the plant design and operation • Knowledge of the peer review process
Documentation	• Describes the peer review team qualifications • Describes the peer review process • Documents where the base PRA does not meet desired characteristics and attributes • Assesses and documents significance of deficiencies • Documents resolutions of deficiencies sufficient to readily allow implementation of resolutions • Describes the scope of the peer review performed (i.e., what was reviewed by the peer review team) • Documents the justification for not reviewing requirements included in the scope of the peer review • Describes any PRA upgrades or NDMs • Identifies the key assumptions (relative to the base PRA) reviewed and the result of the review

C.2.2.2 Peer Review of a Probabilistic Risk Assessment Upgrade or a Newly Developed Method

A peer review should be performed on a PRA upgrade prior to using the upgraded PRA model in support of a PRA application, either for an approved risk-informed program (e.g., risk-informed completion times, 10 CFR 50.69, surveillance frequency control program, NFPA 805) or in the submittal of a risk-informed PRA application for NRC review. The use of an NDM in a PRA is considered a PRA upgrade. Appendix C provides guidance for determining whether a change to the base PRA is a PRA upgrade or PRA maintenance and, if it is a PRA upgrade, whether the change involves the use of an NDM. The incorporation of an NDM results in a PRA upgrade because it involves the application of a method not previously used in the peer-reviewed plant PRA. The staff position for the peer review of a PRA upgrade is discussed in regulatory position C.2.2.2.1.

The peer review of an NDM is performed to assess whether the NDM can be used to support a PRA. The peer review of an NDM focuses on the NDM itself rather than the implementation of the NDM in a PRA. The staff position for the peer review of an NDM is discussed in regulatory position C.2.2.2.2.

C.2.2.2.1 Peer Review of a PRA Upgrade

The peer review of a PRA upgrade is a focused-scope peer review in that it only involves reviewing the changes to the PRA model as a result of the upgrade to the PRA model. The Glossary in this RG provides a definition for PRA upgrade. A PRA upgrade can include the following:

- application of a method not previously used in the peer-reviewed plant PRA,

- implementing a method in a different context, or

- implementing changes that support any SRs previously not reviewed or previously not applicable.

The guidance for the peer review for a PRA upgrade in NEI 17-07, Revision 2, is endorsed by the NRC in this RG.

C.2.2.2.2 Peer Review of a Newly Developed Method

The peer review of an NDM assesses whether the NDM meets a set of technical requirements and, consequently, can be used to support the PRA. Section 5.1 of the PWROG-19027-NP, Revision 2, provides a set of technical requirements for Section 1-6 of the ASME/ANS Level 1/LERF PRA standard. After an NDM has been successfully peer reviewed, the NDM's implementation into a plant-specific PRA is an upgrade and would need an implementation peer review, consistent with the guidance provided in regulatory position C.2.2.2.1. The Glossary of this RG provides a definition for the term NDM and other related terms. Appendix C provides additional discussion about the characteristics and attributes of an NDM.

An acceptable approach to performing a peer review for an NDM is the guidance in NEI 17-07, Revision 2. In particular, NEI 17-07, Revision 2, states, in part, that, if an NDM is deemed not technically acceptable in the NDM peer review report, or if at least one finding-level F&O on the NDM remain open, a licensee or applicant may not use it in a PRA supporting risk-informed licensing applications. If open F&Os from an NDM peer review cannot be successfully closed via an NRC-endorsed peer review process, the NDM could be submitted to the NRC to determine the acceptability of the NDM. Submitted applications that use NDMs with open F&Os related to the NDM will be subject to review by the NRC to determine acceptability of the NDM, its implementation in the PRA, and its potential impact on the application. The peer review of an NDM should meet certain requirements specific to that type of peer review. An acceptable set of requirements against which the adequacy of an NDM can be assessed is provided in PWROG-19027-NP, Revision 2.

Section 5.2 of PWROG-19027-NP, Revision 2, describes features to be added to a peer review report of a plant PRA that uses NDMs and the public availability of NDM peer review reports. Section 5.3 of PWROG-19027-NP, Revision 2, provides the potential outcomes of a peer review of an NDM with respect to whether an NDM is able to support a PRA. NRC endorsement of PWROG-19027-NP, Revision 2, is provided in regulatory positions C.2.2.3 and C.2.2.4 of this RG.

C.2.2.3 Facts and Observation Independent Assessment

The ASME/ANS Level 1/LERF PRA standard requires a peer review of the base PRA against the requirements in that standard. The ASME/ANS Level 1/LERF PRA standard does not require the licensee or applicant to close the F&Os resulting from the peer review of the base PRA prior to using it in applications. However, a licensee or applicant may conduct an F&O independent assessment to close F&Os identified by peer review of the base PRA against a national consensus PRA standard (or from a peer review performed prior to the issuance of RG 1.200, Revision 1, issued January 2007 (Ref. 433)).

One acceptable F&O independent assessment approach is one that is performed according to the NRC-endorsed process, by qualified personnel, and that sufficiently documents the results of the independent assessment. Moreover, in performing an F&O independent assessment, the scope of the independent assessment is identified. The NRC endorses the F&O independent assessment process provided in NEI 17-07, Revision 2, as an acceptable process for meeting regulatory position C.2.2.3.

Furthermore, Section 5.1 of PWROG-19027-NP, Revision 2, provides recommended peer review requirements for Section 1-6 of the ASME/ANS Level 1/LERF PRA standard. The staff finds the recommended peer review requirements to be appropriate for the F&O independent assessment process. For any HLR with one or more SRs that received a finding from the prior peer review, all the applicable SRs under that HLR should be included in the scope of the focused-scope peer review.

The implementation of this staff position is intended to provide the needed confidence in the resolution of peer review F&Os against the requirements in the national consensus PRA standard such that, when used in support of an application, the implementation of the staff position will obviate the need for an in-depth review of the licensee's or applicant's F&O resolutions by NRC reviewers. This allows the staff to focus its review on key assumptions and areas identified by PRA peer reviewers as being of concern and relevant to the application. However, the staff may choose to observe an F&O independent review process or audit an F&O independent assessment closure report.

C.2.2.4 NRC Endorsement of Industry Guidance Documents

NEI 17-07, Revision 2, provides guidance on conducting and documenting a PRA peer review, including peer reviews of NDMs for acceptability and for implementation in a PRA, follow-on peer reviews, and the closure of F&Os that result from a peer review. The NRC staff has reviewed NEI 17-07, Revision 2, against the characteristics and attributes for an acceptable PRA peer review process in regulatory positions C.2.2 through C.2.2.3 with respect to the peer review process for the following:

- a base PRA model,
- a PRA upgrade,
- use of an NDM, and
- independent assessment for the closure of peer review F&Os.

The NRC staff endorses NEI 17-07, Revision 2, as a means of satisfying the peer review requirements in ASME/ANS RA-Sa-2009 and for the purpose of meeting regulatory position C.2 of this RG. The NRC staff has endorsed ASME/ANS RA-Sa-2009, with exceptions and clarifications, in Appendix A to this RG and ASME/ANS RA-S Case 1, with exceptions and clarifications, for seismic PRA in Appendix B to this RG. A peer review can be used to demonstrate that the base Level 1 and limited-Level 2, internal and external hazards, at-power, PRA is acceptable for supporting a risk-informed application if the peer review is conducted consistent with the staff's regulatory position in Appendix A or Appendix B, as applicable, and the guidance in NEI 17-07, Revision 2. Differences between the final published version of ASME/ANS RA-Sa-2009, as endorsed by the NRC, and ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Ref. 444), should be identified and addressed by the licensee in a risk-informed application when ASME RA-Sb-2005 was used to peer-review the licensee's base PRA. In addition, future peer reviews should be performed against the established national consensus PRA standards and peer review guidance, as endorsed in this RG with staff exceptions and clarifications.

As described in regulatory positions C.2.2.2.1 and C.2.2.2.2, the Glossary, and Appendix C to this RG, PWROG-19027-NP, Revision 2, provides definitions of terms, requirements, and guidance related to the peer review of NDMs, and guidance for determining whether a change to a base PRA is PRA maintenance or a PRA upgrade. The staff reviewed PWROG-19027-NP, Revision 2, and endorses the following:

- specific definitions from Section 2, as included and identified in the Glossary of this RG;

- the guidance provided in Section 3 as one acceptable approach for determining whether a change to a PRA model is classified as PRA maintenance or a PRA upgrade, which is consistent with regulatory position C.2.2.2 and Appendix C to this RG;

- the recommended Section 1-7 on NDMs for the ASME/ANS Level 1/LERF PRA standard as documented in Section 4, which is consistent with regulatory position C.2.2.2.2 of this RG;

- the recommended requirements in Section 5.1, which are consistent with regulatory position C.2.2.2.1 of this RG, with the following clarification:

    - PWROG-19027-NP, Revision 2, states that the extent of a focused-scope peer review includes all SRs under an HLR with an SR that received a peer review finding. However, the staff finds that, for a focused-scope peer review, the scope of the focused-scope peer review should include all applicable SRs for any HLR with at least one SR that received a finding from the prior peer review;

- the features described in Section 5.2 that are to be added to a peer review report of a plant PRA that uses NDMs and the public availability of NDM peer review reports; and

- the potential outcomes of a peer review of an NDM described in Section 5.3 with respect to whether an NDM is able to support a PRA, which meets regulatory position C.2.2.2.2 of this RG.

NEI 17-07, Revision 2, provides guidance on peer review for the acceptability of an NDM and peer review for the implementation of an NDM in a PRA and draws a distinction between these two types of peer reviews for NDMs.

The acceptability peer review of an NDM determines whether the NDM is suitable to be used to support a PRA and should be assessed against the NRC-endorsed requirements for NDMs from PWROG-19027-NP, Revision 2, as discussed above. The peer review team members for the NDM acceptability peer review should possess experience and expertise sufficient to allow for a comprehensive understanding of the NDM. Unlike the peer review for the implementation of methods, the acceptability peer review of NDMs involves a detailed examination of supporting information beyond a sampling review. Furthermore, all finding-level F&Os should be closed using an NRC-endorsed process before the NDM is used to support an application. A summary report of the NDM acceptability peer review should be documented and submitted to the NRC, which details the results of the peer review and provides, among other things, conclusions regarding the acceptability of the NDM.

Regardless of whether an acceptability or implementation peer review for an NDM is performed concurrent with or independent of a peer review of a base PRA, the guidance in NEI 17-07, Revision 2, as endorsed in this RG, provides acceptable guidance for conducting the peer review.

C.3 Demonstrating the Acceptability of a Probabilistic Risk Assessment Used to Support an Application

This section of the RG provides guidance to licensees and applicants on an approach acceptable to the NRC staff to demonstrate the acceptability of the base PRA used, in total or in part, to support an application. PRA acceptability for a given risk-informed activity is determined in the context of the staff position in this RG, relevant application-specific regulatory guidance, and the related requirements (e.g., license condition) for the application.

The application-specific RGs identify the specific PRA results used to support the decisionmaking and the portions of the base PRA needed to provide those results. The portions of the base PRA needed to support the application-specific analysis should be identified, and the guidance in this RG applies to those portions. Regulatory positions C.3.1 and C.3.2 summarize the staff position on the outcome of the application of the application-specific RGs in determining the scope of application of this RG. One acceptable approach to demonstrate conformance with regulatory positions C.3.1 and C.3.2 is to use a national consensus standard. The ASME/ANS Level 1/LERF PRA standard provides the technical requirements for this purpose. If the ASME/ANS Level 1/LERF PRA standard is implemented, as endorsed by the staff in Appendices A and B to this RG, as applicable, with exceptions and clarifications, regulatory positions C.3.1 and C.3.2 are considered to be met.

When using this RG, it is anticipated that the licensee's or applicant's description of the application should include the following:

- SSCs, operator actions, and plant operational characteristics affected by the application;
- a description of the cause-effect relationships related to the change and the affected SSCs, operator actions, and plant operational characteristics;
- mapping of the cause-effect relationships onto PRA model elements;
- identification of the PRA results that will be used to compare against the applicable acceptance criteria or guidelines and how the comparison is to be made; and
- the scope of risk contributors (hazard groups and modes of operation) included in the PRA to support the decision.

C.3.1 Scope of Risk Contributors Addressed by the Probabilistic Risk Assessment Model

Based on the definition in the application, and in particular the acceptance criteria or guidelines, the scope of risk contributors (internal and external hazards and plant operating modes) for the PRA should be identified. For example, if the application is designed around using the acceptance guidelines of RG 1.174, the evaluations of CDF, the change in CDF (i.e., ΔCDF), LERF, and the change in LERF (i.e., ΔLERF) should be performed with a full-scope PRA, including all hazard groups and all modes of operation. However, since many PRAs are not full scope and do not include models for all risk contributors, decisionmakers should allow for such omissions. Examples of allowances that may be made for omitted PRA scope items include using compensatory measures, restricting the implementation of the proposed change to those aspects of the plant covered by the risk model, and using bounding analyses to show the risk contribution is likely not greater than a certain value. However, it should be noted that, consistent with the Commission-endorsed phased PRA quality initiative, all risk contributors that cannot be shown to be insignificant to the decision should be assessed through quantitative risk assessment methods to support risk-informed licensing actions. This RG is focused specifically on the acceptability of the PRA information used in support of a decision and does not address the acceptability of non-PRA information used to justify the omission of PRA scope items.

The national consensus PRA standards and industry PRA programs that have been developed address a specific scope. For example, the ASME/ANS Level 1/LERF PRA standard and NEI 17-07, Revision 2, addresses a Level 1 and a limited Level 2 PRA analysis for all hazards (i.e., internal events, internal flood, internal fire, seismic, high wind, external flood, and other hazards), with respect to at-power operations.

C.3.2 Identification of the Parts of a Probabilistic Risk Assessment Used to Support the Application

Based on an understanding of how the PRA model is to be used to achieve the desired results, the licensee or applicant should have identified the portions of the PRA for each hazard group required to support a specific application. This includes the following two categories of items: (1) the PRA logic model elements onto which the cause-effect relationships are mapped (i.e., those directly affected by the application) and (2) all the events with mapped cause-effect relationships that appear in the accident sequences. For some applications, this may be some subset of all items in the base PRA, but for others (e.g., risk-informing the scope of special treatment requirements), all portions of the PRA model may be relevant.

C.3.3 Demonstration of Probabilistic Risk Assessment Acceptability

There are two aspects to demonstrating the acceptability of the portions of the PRA used to support an application. The first aspect is the assurance that the portions of the PRA used in the application have been developed and performed in a technically correct manner. The second aspect is the assurance that the assumptions and approximations used in developing the PRA are appropriate.

For the first aspect, assurance that the portions of the PRA used in the application have been developed and performed in a technically correct manner indicates that (1) the PRA model, or those portions of the model required to support the application, represents the as-designed or as-built and as-operated plant, which, in turn, indicates that the PRA reflects the current design and operating practices and experience, where appropriate, (2) the PRA logic model has been developed in a manner consistent with industry good practice (see the Glossary for this RG) and that it correctly reflects the dependencies amongst systems, components, and operator actions, and (3) the probabilities and frequencies used are estimated consistently with the definitions of the corresponding events in the PRA logic model. The NRC recognizes there may be special circumstances where credit for planned modification(s) in the PRA model is appropriate, and these applications are addressed by the staff on a case-by-case basis. Regarding the peer review of a base PRA model that credits planned modifications, licensees or applicants should provide details of the planned modifications to the peer review team in advance of the peer review. These details should include any spatial information associated with the planned modification that may impact, for example, internal flood, internal fire, seismic, high-wind, or external flood PRA, or other PRAs of hazards that are dependent on spatial information. The resulting peer review report should clearly identify any planned modifications reviewed by the peer review team. Regulatory position C.4.2 provides guidance on submittal documentation for such cases.

For the second aspect, the current state-of-practice in PRA technology is that there are issues for which there is no consensus on methods of analysis. Furthermore, PRAs are models that rely on certain approximations and judgments to make the models tractable and certain assumptions that address uncertainties related to modeling specific issues. Regulatory position C.3.3.2 of this RG, RG 1.174, and NUREG-1855 provide guidance on how to address and treat the uncertainties associated with a PRA. In accordance with that guidance, the impact of these assumptions and approximations on the results of interest to the application should be understood.

C.3.3.1 Assessment that the Probabilistic Risk Assessment Model Is Technically Correct

When using risk insights based on a PRA model, the licensee or the applicant ensures that the PRA model, or at least those portions of it needed to provide the results, is technically correct as discussed above.

The licensee is to demonstrate that the PRA model represents the current plant design and configuration and represents current operating practices and operating experience to the extent required to support the application. This demonstration can be achieved through a PRA configuration control plan that includes provisions for updating the model periodically to reflect changes that impact the significant accident sequences.

The various national consensus PRA standards and industry documents that provide guidance on the development, performance, and peer reviews of PRAs are discussed, in regulatory positions C.2 through C.2.2.4 of this RG. Details of the staff's clarifications and qualifications that support regulatory positions are provided in Appendices A, B, and C.

When the exceptions and clarifications raised by the staff are taken into account, the national consensus PRA standard or PRA peer review process in question is considered to be acceptable for the purpose for which it was intended. If the portions of the PRA can be shown to have met the requirements of these documents, with attention paid to the NRC's objections, it can be assumed that the analysis is technically correct. Therefore, the staff should be able to focus on the assumptions and approximations relevant to the application (as discussed below in regulatory position C.3.3.2), and, other than an audit, a detailed review by NRC staff of the base model PRA should not be necessary. When deviations from these documents exist, the applicant should demonstrate either that its approach is equivalent or that the influence on the results used in the application are such that no changes occur in the significant accident sequences or contributors.

C.3.3.2 Assessment of Assumptions and Approximations

Since the standards and industry PRA programs are not (or are not expected to be) prescriptive, there is some freedom on how to model certain phenomena or processes in the PRA; different analysts may make different assumptions and still be consistent with the requirements of the standard, or the assumptions may be acceptable under the guidelines of the peer review process. The choice of a specific assumption or a particular approximation may influence the results of the PRA. For each application that calls upon this RG, the assumptions and approximations relevant to that application and those that are key to that application (see the definitions of "assumption," "key assumption," and "key source of uncertainty" in the Glossary of this RG) are identified. The key assumptions for a PRA application are generally identified from the assumptions and approximations identified in the base PRA. The identified key assumptions are used to identify sensitivity studies as input to the decisionmaking associated with the application. When a key assumption is shown to be consistent with a consensus method or approach, that key assumption may no longer be subject to additional sensitivity studies in the context of a PRA application. Appendices A, B, and C to this RG either identify the need for or directly relate to (in the case of the industry peer review process) a peer review. One of the functions of the peer review is to assess the assumptions and make judgments as to their appropriateness.

C.4 Documentation to Support a Regulatory Decision

The licensee or applicant develops documentation of the PRA model and the analyses performed to support the risk-informed regulatory activity. This documentation comprises both archival (i.e., available for audit or inspection) and submittal (i.e., submitted as part of the risk-informed request) documentation. The former may be required on an as-needed basis to facilitate the NRC staff's review of the risk-informed submittal.

The results of a PRA are sometimes used to support a decision changing a regulatory activity that is made by the licensee as authorized by a previous licensing decision. Generally, all such authorizations have an associated PRA configuration control process that is part of the amendment authorizing the use of a PRA. The licensee follows the PRA configuration control process for the approved application. The following guidance for archival documentation is associated with the base PRA and is not affected by any application-specific guidance.

C.4.1 Archival Documentation

Archival documentation associated with the base PRA includes the following:

- Process to determine acceptability: A detailed description of the process used to determine the acceptability of the PRA is provided.

- Peer reviews: Includes the results of the peer review of the base PRA, PRA upgrades, and use of NDMs, and the results of the F&O independent assessment (as discussed in regulatory position C.2.2). Descriptions of the resolution of all the peer reviews (i.e., base PRA, PRA upgrades, and use of NDMs) and the F&O independent assessment are included. The results are documented in such a manner that it is clear why each requirement is considered to have been met. This can be done, for example, by providing a reference to the appropriate section of the PRA model documentation.

- Base PRA model: Regulatory position C.1.3 of this guide provides the characteristics and attributes of archival documentation associated with the base PRA. The documentation maintained by the licensee or applicant should be legible, retrievable (i.e., traceable), and of sufficient detail that the staff can comprehend the bases supporting the results used in the application to support potential audits and inspections by the NRC staff.

- Maintenance and upgrades: A description of the process for maintenance, upgrade, and use of NDMs is provided. The history is maintained of the activities related to PRA maintenance, PRA upgrades, and use of NDMs. The history includes the results of peer reviews that were performed as a result of a PRA upgrade or the use of an NDM.

The archival documentation associated with a specific application includes enough information to demonstrate that the scope of the review of the base PRA is sufficient to support the application. This includes the following information:

- the impact of the application on the plant design, configuration, or operational practices;

- the risk assessment, including a description of the methodology used to assess the risk of the application, how the base PRA model was modified to appropriately model the risk impact of the application, and details of quantification and the results;

- the acceptance guidelines and method of comparison;

- the scope of the risk assessment in terms of hazard groups and specific accident scenarios and operating modes modeled;

- the portions of the PRA required to provide the results needed to support comparison with the acceptance guidelines; and

- documentation associated with NDMs (e.g., detailed descriptions of the NDM, assumptions, scope, limitations, data used along with the bases for data selection, technical bases, equations) developed or sponsored by the licensee or the applicant. Documentation associated with the implementation of the NDM (e.g., self-assessment reports, peer-review reports including the disposition of findings, independent assessment team closure report) that have been incorporated into the licensee's or applicant's PRA model. If the licensee or the applicant has used an NDM developed by a vendor or a different licensee or applicant, information on that NDM received from that vendor or licensee or the applicant should be included as part of the archival documentation associated with a specific application.

C.4.2 Licensee or Applicant Submittal Documentation

For applications that require prior NRC approval, demonstration that the acceptability of the PRA used in an application is sufficient may be done through the submittal of the following information to the NRC. Previously submitted documentation may be referenced if it is acceptable for the subject submittal:

- information to assure that the PRA model represents the as-designed or as-built and as-operated plant:

  - Identify permanent plant changes (such as design or operational practices) that have an impact on those things modeled in the PRA but have not been incorporated in the base PRA model. If a plant change has not been incorporated, provide a justification of why the change does not impact the PRA results used to support the application. This justification should be in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application decision were not adversely impacted or remained the same.

- documentation that the portions of the PRA required to produce the results used in the decision are performed consistently with the standard as endorsed with exceptions and clarifications in the appendices to this RG. If a requirement of the standard (as endorsed with exceptions and clarifications in the Appendix to this guide) has not been met, provide a justification of why it is acceptable that the requirement has not been met. This justification should be in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application were not adversely impacted or remained the same.

- a summary of the risk assessment methodology used to assess the risk of the application, including how the base PRA model was modified to appropriately model the risk impact of the application and results (note that this is the same as that identified in the application-specific RGs).

- identification of the key assumptions and approximations relevant to the results used in the decision-making process, as well as the peer reviewers' assessment of those assumptions. These assessments provide information to the NRC staff in its determination of whether the use of these assumptions and approximations is appropriate for the application, or whether sensitivity studies performed to support the decision are appropriate. A peer review of planned modifications should clearly identify and describe the plant modifications and design changes that are modeled in the PRA but are not completed at the time of the licensing application submittal.

- discussion, in the following forms, of the resolution of the peer review findings and observations that are applicable to the portions of the PRA required for the application:

  - a discussion of how the PRA model has been changed, and

  - a justification in the form of a sensitivity study that demonstrates the accident sequences or contributors significant to the application decision were not adversely impacted or remained the same as a result of the particular issue

- identification of the use of the portions of the PRA that conform to capability categories or grades lower than deemed required for the given application (See Section 1-3 of ASME/ANS RA-Sa-2009) if the standards or peer review process documents recognize different capability categories or grades that are related to level of detail, degree of plant specificity, and degree of realism

- identification of the PRA model upgrades, including the use of NDMs. The results of the peer review of these PRA upgrades, including those for the implementation of NDMs, should be submitted to the staff. If applicable, identify NDM documentation previously submitted to the NRC consistent with the guidance in NEI 17-07, Revision 2, as well as PWROG-19027-NP, Revision 2, and the regulatory position C.2.2.4 of this RG.

- discussion of the resolution of the peer review findings for the NDMs if the licensee's or applicant's PRA model includes NDMs that have open finding-level F&Os from the technical assessment peer review against the NDM criteria as endorsed in regulatory positions C.2.2.2.1 and C.2.2.2.2 of this RG

- documentation associated with NDMs to support a review of the technical acceptability of the NDM by the NRC staff if the licensee's or applicant's PRA model includes NDMs that have not been subjected to the technical assessment peer review against the NDM criteria as endorsed in regulatory positions C.2.2.2.1 and C.2.2.2.2 of this RG. Such documentation should include, for example, detailed descriptions of the NDM, assumptions, scope, limitations, data used along with the bases for data selection, technical bases, and equations developed or sponsored by the licensee or the applicant. Documentation associated with the implementation of the NDM (e.g., self-assessment reports, peer review reports including the disposition of findings, independent assessment team closure report) that has been incorporated into the licensee's or applicant's PRA model also should be submitted.

D. IMPLEMENTATION

The NRC staff may use this regulatory guide as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this regulatory guide to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, “Backfitting,” and as described in NRC Management Directive 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests” (Ref. 45), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants.” The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in Management Directive 8.4. If a licensee believes that the NRC is using this regulatory guide in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

GLOSSARY

as-built and as-operated	The representation in a PRA model of the current plant design, configuration, procedures, and performance data (e.g., component failure rates).
as-designed, as-to-be-built, and as-to-be-operated	The term “as-designed” refers to the PRA used to model the plant configuration in the DC or COL stage, in which the plant is not yet built or operated and, therefore, this PRA reflects the plant as it is intended to be built (i.e., as-to-be-built) and as it is intended to be operated (i.e., as-to-be-operated).[6]
assumption	A decision or judgment that is made in the development of a model, implementation of a method, or conduct of an analysis in development of a PRA model for modeling convenience or because of a lack of information or state-of-knowledge. An assumption is considered to be credible when it has a sound technical basis and the technical basis would receive broad acceptance within the relevant technical community. However, an assumption may be related to scope or level of detail and is made for modeling convenience in the knowledge that a more detailed model would produce different results.
base PRA	The PRA from which results or insights are derived or that is modified and/or manipulated to support a risk-informed NRC regulatory activity. In some cases, such as applications related to 10 CFR 50.69, the PRA used in the application may be the base PRA. The base PRA provides a quantitative assessment of the identified risk of the as-built and as-operated plant in terms of scenarios that result in undesired consequences (e.g., core damage, large early release, or a large release) and their frequencies, and it comprises specific technical elements in performing the quantification. The base PRA serves as the foundational representation of the as-built and as-operated plant necessary to support an application.
consensus method/model	In the context of risk-informed regulatory decisions, a method or model approach that the NRC has used or accepted for the specific risk-informed application for which it is proposed. A consensus method or model may also have a publicly available, published basis and may have been peer reviewed and widely adopted by an appropriate stakeholder group.
conservative	Relates to the use of information (e.g., assumptions) such that the assessed outcome is meant to be less realistic in a cautious manner as compared to the expected outcome.

6 The NRC recognizes there may be special circumstances where credit for planned modification(s) in the PRA model is appropriate and these applications are addressed by the staff on a case-by-case basis. Regarding the peer review of a base PRA model that credits planned modifications, licensees or applicants should provide details of the planned modifications to the peer review team in advance of the peer review. These details should include any spatial information associated with the planned modification that may impact, for example, internal flood, internal fire, seismic, high-wind, or external flood PRA, or other PRAs of hazards that are dependent on spatial information. The resulting peer review report should clearly identify any planned modifications reviewed by the peer review team. Regulatory position C.4.2 provides guidance on submittal documentation for such cases.

current good practice (or state-of-practice)	Practices that are widely accepted by and implemented throughout the commercial nuclear power industry, have been shown to be technically acceptable in well-documented analyses or engineering assessments that are publicly available, and are accepted by the NRC.
demonstrably conservative	Relates to the use of information (e.g., assumptions) that provides high confidence that the assessed outcome is as conservative as it is portrayed to be.
key assumption	An assumption for a risk-informed decision that could affect the PRA results that are being used in a decision and, consequently, may influence the decision being made. An assumption may be a key assumption relative to the base PRA or relative to an application. With respect to a base PRA, an assumption would be considered key if it affects the insights gained from the PRA results.
key source of uncertainty	A key source of uncertainty relates to an issue for which there is no consensus approach or model and where the choice of approach or model is known to have an impact on the risk profile (e.g., total CDF and total LERF (or total LRF, as applicable)), the set of initiating events and accident sequences that contribute most to CDF and LERF (or LRF, as applicable) such that it influences a decision being made using the PRA. Such an impact might occur, for example, by introducing a new functional accident sequence or a change to the overall CDF or LERF (or LRF, as applicable) estimates significant enough to affect insights gained from the PRA.
level of detail	Relates to the degree to which (i.e., amount of) information is discretized and included in the model or analysis.
model[7]	A qualitative and/or quantitative representation that is constructed to portray the inherent characteristics and properties of what is being represented (e.g., a system, component or human performance, theory or phenomenon). A model may be in the form, for example, of a structure, schematic or equation. Method(s) are used to construct the model under consideration.
newly developed method[7]	A PRA method that has either been developed separately from a state-of-practice method or is one that involves a fundamental change to a state-of-practice method. An NDM is not a state-of practice or a consensus method.
PRA	An approach that (1) provides a quantitative assessment of the identified risk in terms of scenarios that result in undesired consequences (e.g., core damage or a large early release) and their frequencies and (2) comprises specific technical elements in performing the quantification.
PRA acceptability	Measured in terms of PRA scope, the level of detail in the PRA, the PRA's conformance with the PRA technical elements in regulatory position C.1.2 of this RG, and how closely the PRA represents a plant's actual configuration and operations. PRA acceptability is determined for each risk-informed activity considering the staff positions in this RG, staff positions in relevant

7 These terms and their definitions are reproduced from PWROG-19027-NP, Revision 2.

application-specific regulatory guidance, and any related requirements (e.g., license conditions) for the application.

PRA application	A documented analysis based in part or whole on a plant-specific PRA that is used to assist in decisionmaking with regard to the design, licensing, procurement, construction, operation, or maintenance of a nuclear power plant.
PRA maintenance[7]	A change in the PRA that does not meet the definition of PRA upgrade.
PRA method	An analytical approach in the PRA used to satisfy a supporting requirement or collection thereof from a national consensus PRA standard. An analytical approach is generally a compilation of the analyses, tools, assumptions, and data used to develop a model. For example, application of the MELCOR code, a system-level code that contains engineering analyses and assumptions supported by experimental data, is the method used to develop a response model that predicts the performance of the core during PRA transients.
PRA upgrade[7]	A change in the PRA that results in the applicability of one or more supporting requirements that were not previously included within the PRA (e.g., performing qualitative screening for Part 4 of ASME/ANS Level 1/LERF PRA standard when the related high-level requirement was previously not applicable, or adding a new hazard model), an implementation of a PRA method in a different context, or the incorporation of a PRA method not previously used.
realism	an accurate representation (to the extent practical) that reflects the expected response of the as-built and as-operated plant.
risk significance	Design or operational features of the plant, including operator actions, that are important contributors because of their ability to either increase or decrease the risk. With regard to a risk significant contributor (e.g., risk-significant accident sequence, risk significant basic event, risk significant HFE), significance (or contribution) is measured with respect to the degree to which the contributor impacts the decision under consideration. For the base PRA model, significance can be measured with respect to the contribution to the total CDF or LERF (or LRF, as applicable), or it can be measured with respect to the contribution to the CDF or LERF (or LRF, as applicable) for a specific hazard group or POS, depending on the context. For example, for the purposes of defining capability categories, the ASME/ANS Level 1/LERF PRA standard defines significance at the hazard group level. Whatever the context, the numerical criteria in the definitions of "significant accident sequence" and "significant basic event/contributor" are recommended.
significant accident sequence	An accident sequence that is part of the set of sequences that, when ranked by risk contribution from highest to lowest, compose 95 percent of the CDF or the LERF (or LRF, as applicable) for the hazard group under consideration, or that individually contribute more than 1 percent to the CDF or LERF (or LRF, as applicable) for the hazard group under consideration. This set of

risk-significant accident sequences may be defined functionally or in terms of systems.

significant basic event/contributor	A basic event (i.e., equipment unavailabilities and HFEs) or risk contributor whose Fussell-Vesely importance measure value is greater than 0.005 or the risk-achievement worth importance measure value is greater than 2.
state-of-practice	See the definition for "current good practice."

REFERENCES [8]

The References section applies to versions of the documents available at the time of this RG's issuance. Licensees or applicants using this RG should check all referenced documents to ensure no change has occurred since issuance of the RG.

1. U.S. Nuclear Regulatory Commission (NRC), "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," *Federal Register*, Volume 60, No. 158: pp. 42622, (60 FR 42622), Washington, DC, August 16, 1995.

2. *U.S. Code of Federal Regulations* (CFR), "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter 1, Title 10, "Energy."

3. CFR, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter 1, Title 10, "Energy."

4. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Washington, DC.

5. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 19.0, "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," Washington, DC.

6. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Washington, DC.

7. NRC, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," Washington, DC.

8. NRC, NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," Washington, DC.

9. NRC, RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Washington, DC.

10. NRC, RG 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," Washington, DC.

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8 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

11. NRC, RG 1.177, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," Washington, DC.

12. NRC, RG 1.178, "An Approach for Plant-Specific, Risk-Informed Decisionmaking for Inservice Inspection of Piping," Washington, DC.

13. NRC, RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," Washington, DC.

14. NRC, RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Washington, DC.

15. National Fire Protection Association (NFPA), Standard 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants," Quincy, MA.[9]

16. NRC, RG 1.206, "Applications for Nuclear Power Plants," Washington, DC.

17. NRC, DC/COL-ISG-020, "Interim Staff Guidance on Implementation of a Probabilistic Risk Assessment-Based Seismic Margins Analysis for New Reactors," Washington, DC, March 15, 2010.

18. NRC, SRM-SECY-93-087, "SECY-93-087—Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light Water Reactor (ALWR) Designs," Washington, DC, July 21, 1993. (ADAMS Accession No. ML003708056)

19. NRC, DC/COL-ISG-028, "Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application," Washington, DC, November 2016.

20. American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RA-S-2008, ASME, New York, NY, ANS, La Grange Park, Illinois, February 2009.

21. Nuclear Energy Institute (NEI), NEI 17-07, Revision 2, "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard," Washington, DC, August 2019. (ADAMS Accession No. ML19241A615)

22. ASME/ANS Standard ASME/ANS RA-S Case 1, "Case for ASME/ANS RA-Sb-2013 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment of Nuclear Power Plant Applications," ASME, New York, NY, November 22, 2017.[10]

9 The NFPA makes important safety codes and standards available for free online and documents are available at http://www.nfpa.org/codes-and-standards/document-information-pages. They may also be purchased by calling NFPA Customer Sales 800.344.3555 or writing NFPA 1 Batterymarch Park, Quincy, MA 02169-7471.

10 Copies of ASME standards may be purchased from ASME, Two Park Avenue, New York, New York 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web-based store at http://www.asme.org/Codes/Publications/.

23. Pressurized Water Reactor Owners Group (PWROG), Report PWROG-19027-NP, Revision 2, “Newly Developed Method Requirements and Peer Review,” Cranberry Township, PA, July 2020. (ADAMS Accession No. ML20213C660)

24. NRC, RG 1.200, Revision 2, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Washington, DC, March 2009. (ADAMS Accession No. ML090410014)

25. NRC, SECY-00-0162, “Addressing PRA Quality In Risk-Informed Activities,” Washington, DC, July 28, 2000. (ADAMS Accession No. ML003732744)

26. NRC, SECY-04-0118, “Plan for the Implementation of the Commission’s Phased Approach to Probabilistic Risk Assessment Quality,” Washington, DC, July 13, 2004. (ADAMS Accession No. ML041470505)

27. NRC, SECY-07-0042, “Status of the Plan for the Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality,” Washington, DC, March 7, 2007. (ADAMS Accession No. ML063630346)

28. NEI 00-02, Revision A3, “Probabilistic Risk Assessment (PRA) Peer Review Process Guidance,” Washington, DC, March 20, 2000. (ADAMS Accession No. ML003728023)

29. NEI 05-04, Revision 2, “Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard,” Washington, DC, November 2008. (ADAMS Accession No. ML083430462)

30. NEI 07-12, Revision 0, Draft H, “Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines,” Washington, DC, November 2008. (ADAMS Accession No. ML083430464)

31. NEI 12-13, “External Hazards PRA Peer Review Process Guidelines,” Washington, DC, August 2012. (ADAMS Accession No. ML12240A027)

32. Franovich, M. X., NRC, letter to Krueger, G., NEI, “U.S. Nuclear Regulatory Commission Acceptance of Nuclear Energy Institute (NEI) Guidance NEI 12-13, ‘External Hazards PRA Peer Review Process Guidelines’” (August 2012),” Washington, DC, March 7, 2018. (ADAMS Accession No. ML18025C025)

33. NEI, "NEI 05-04/07-12/12-06 Appendix X: 'Close-Out of Facts and Observations (F&Os)’,” Washington, DC, February 21, 2017. (ADAMS Package Accession No. ML17086A431)

34. Giitter, J. G., and Ross-Lee, M. J., NRC, letter to Krueger, G., NEI, “U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," Washington, DC, May 3, 2017. (ADAMS Accession No. ML17079A427)

35. NRC, Management Directive 6.6, "Regulatory Guides," Washington, DC, May 2, 2016. (ADAMS Accession No. ML18073A170)

36. International Atomic Energy Agency (IAEA) Safety Standard Series No. SSG-3, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants," Vienna, Austria, 2010.[11]

37. IAEA Safety Standard Series No. SSG-4, "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants," Vienna, Austria, 2010.

38. NRC, SRM-SECY-90-016, "SECY-90-16—Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," Washington, DC, June 26, 1990. (ADAMS Accession No. ML003707885)

39. NRC, SECY-13-0029, "History of the Use and Consideration of the Large Release Frequency Metric by the U.S. Nuclear Regulatory Commission," Washington, DC, March 22, 2013. (ADAMS Accession No. ML13022A207)

40. NRC, SRM-SECY-12-0081, "Staff Requirements—SECY-12-0081 – Risk-Informed Regulatory Framework for New Reactors," Washington, DC, October 22, 2012. (ADAMS Accession No. ML12296A158)

41. NRC, NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," Washington, DC, April 2005. (ADAMS Accession No. ML051160213)

42. NRC, NUREG-1842, "Evaluation of Human Reliability Analysis Methods Against Good Practices," Washington, DC, September 2006. (ADAMS Accession No. ML063200058)

43. NRC, RG 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Washington, DC, January 2007. (ADAMS Accession No. ML070240001)

44. ASME Standard ASME RA-Sb-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," New York, NY, December 30, 2005.

45. NRC, Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection," Washington, DC.

11 Copies of International Atomic Energy Agency (IAEA) documents may be obtained through their Web site: www.iaea.org/ or by writing the International Atomic Energy Agency, P.O. Box 100 Wagramer Strasse 5, A-1400 Vienna, Austria.

APPENDIX A

NRC REGULATORY POSITION ON ASME/ANS RA-SA-2009

A-1 Introduction

The American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) previously published ASME/ANS RA-Sa-2009, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Ref. 1). The standard states that it "sets forth requirements for probabilistic risk assessments (PRAs) used to support risk-informed decisions for commercial nuclear power plants, and describes a method for applying these requirements for specific applications." The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed ASME/ANS RA-Sa-2009 against the characteristics and attributes of an acceptable PRA as discussed in regulatory positions C.1 and C.2 of this regulatory guide (RG). The staff's position on each requirement (referred to in the standard as a requirement, a high-level requirement, or a supporting requirement) in ASME/ANS RA-Sa-2009 is categorized as "no objection," "no objection with clarification," or "no objection subject to the following qualification," and defined as follows:

- **No objection**. The staff has no objection to the requirement.

- **No objection with clarification**. The staff has no objection to the requirement. However, certain requirements, as written, are either unclear or ambiguous, and therefore the staff has provided its understanding of these requirements.

- **No objection subject to the following qualification**. The staff has a technical concern with the requirement and has provided a qualification to resolve the concern.

ASME/ANS RA-Sa-2009 PRA standard is divided into 10 parts:

- Part 1 — general requirements
- Part 2 — technical and peer review requirements for internal events
- Part 3 — technical and peer review requirements for internal flood
- Part 4 — technical and peer review requirements for internal fire events
- Part 5 — technical and peer review requirements for seismic events
- Part 6 — technical and peer review requirements for screening of other external hazards
- Part 7 — technical and peer review requirements for high wind
- Part 8 — technical and peer review requirements for external flood
- Part 9 — technical and peer review requirements for other external hazards
- Part 10 — technical and peer review requirements for seismic margins

Tables A-1 through A-10 provides the staff's position on each requirement in Parts 1 through 10, respectively. A discussion of the staff's concern (issue) and the staff proposed resolution is provided. In the proposed staff resolution, the staff clarification or qualification to the requirement is indicated in either bolded text (i.e., **bold**) or strikeout text (i.e., ~~strikeout~~); that is, the necessary additions or deletions to the requirement (as written in the ASME/ANS Level 1/LERF PRA standard) have been provided for the staff to have no objection.

Table A-1. Staff Position on ASME/ANS RA-Sa-2009 Part 1, General Requirements for an At-Power Level 1 and LERF PRA

Index No	Issue	Position	Resolution
Global			
References	Use of references: the various references, may be acceptable, in general; however, the staff has not reviewed the references, and there may be aspects that are not applicable or not acceptable.	Clarification	For every reference cited in the standard (except NEI 00-02): **No staff position is provided on this reference. The staff neither approves nor disapproves of information contained in the referenced document.**
Section 1-1			
1-1.1 thru 1-1.7	--------------------	No objection	--------------------
Section 1-2			
1-2.1	Acronyms		
COL	Acronym is needed	Clarification	***COL*: Combined License**
Other acronyms	--------------------	No objection	--------------------
1-2.2	Definitions		
Definitions	The NRC has developed a glossary of terms in this RG, which includes definitions of some terms that were derived from PWROG-19027-NP, Revision 2, and that were previously defined in ASME/ANS RA-Sa-2009.	Clarification	The NRC endorses the definitions of the following terms, as presented in the glossary of this RG: • as-built, as-operated • assumption • demonstrably conservative • PRA maintenance • PRA upgrade • significant accident sequence • significant basic event/contributor All other terms and definitions in Section1-2.2 of ASME/ANS RA-Sa-2009 are endorsed with no objection.
Section 1-3			
1-3.1 thru 1-3.4, 1-3.6	--------------------	No objection	--------------------
1-3.5, 2nd paragraph	Use of the word "significant" should match definitions provided in Section 2.2.	Clarification	(b) The difference is not significant if the modeled accident sequences accounting for at least ~~90%~~ **95%** of CDF/LERF for the hazard group ….

**Table A-1. Staff Position on ASME/ANS RA-Sa-2009 Part 1,
General Requirements for an At-Power Level 1 and LERF PRA**

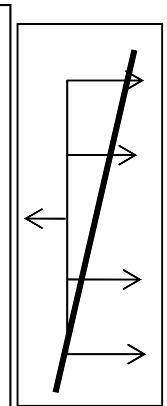
Index No	Issue	Position	Resolution
Figure 1-3-1	See staff proposed resolution for Section 1-1.4.2, text in Box 4 of Figure 1-3.1-1 needs to be modified be consistent with the text.	Clarification	For each relevant Hazard Group, ↓ 4 Determine **the relative importance to the application, identify the portions of the HG PRA relevant to the application, and for each relevant portion of the hazard group, determine the** Capability Category **for each SR** needed for each potion of PRA to  ↓ 5 PRA scope and risk metrics sufficient to evaluate plant ↓
Section 1-4			
1-4.1 thru 1-4.3.2, 1-4.3.4 thru 1-4.5	--------------------	No objection	--------------------
1-4.3.3, 2nd paragraph	The intent of this statement/requirement is for the use of outside expert, as such the use of the word “should” does not provide a minimum requirement.	Clarification	…The PRA analysis team **shall** ~~should~~ use outside experts, even when….
Section 1-5			
1-5.1 thru 1-5.7	--------------------	No objection	--------------------

Table A-1. Staff Position on ASME/ANS RA-Sa-2009 Part 1, General Requirements for an At-Power Level 1 and LERF PRA

Index No	Issue	Position	Resolution
Section 1-6			
1-6.1.1, 1-6.1.2, 1-6.2, 1-6.4, 1-6.5, 1-6.6.2	-------------------	No objection	-------------------
1-6.1	The purpose, as written, implies that it is solely an audit against the requirements of Section 4. A key objective of the peer review is to ensure when evaluating the PRA against the technical requirements, the "quality" (i.e., strengths and weaknesses) of the PRA; this goal is to be clearly understood by the peer review team. Further, the statement that "the peer review need not assess all aspects of the PRA against all requirements" could be taken to imply that some of the requirements could be skipped.	Clarification	…another purpose of the peer review is to determine the strengths and weaknesses in the PRA. **Therefore, the peer review shall also assess the appropriateness of the assumptions.** The peer review need not assess all aspects of the PRA against all requirements in the Technical Requirements Section of each respective Part of this Standard; however, enough aspects of the PRA shall be reviewed for the reviewers to achieve consensus on the **assessment of each applicable supporting requirement, as well as on the** adequacy of methodologies and their implementation for each PRA Element.
1-6.3	As written, there does not appear to be a minimum set. The requirement as written provides "suggestions." A minimal set of items is to be provided; the peer reviewers have flexibility in deciding on the scope and level of detail for each of the minimal items.	Clarification	The peer review team shall use the requirements… of this Standard. **For each PRA element, a set of review topics required for the peer review team are provided in the subparagraphs of para. 6.3.** Additional material for those Elements may be reviewed depending on the results obtained. ~~These suggestions are not intended to be a minimum or comprehensive list of requirements.~~ The judgment of the reviewer shall be used to determine the specific scope and depth ~~of the review in each~~ **of each review topic for each** PRA element.

Table A-1. Staff Position on ASME/ANS RA-Sa-2009 Part 1, General Requirements for an At-Power Level 1 and LERF PRA

Index No	Issue	Position	Resolution
1-6.6.1	The specific SRs addressed in the peer review need to be documented. As written it is not clear whether certain essential items are included in the documentation requirements that are necessary to accomplish the goal of the peer review.	Clarification	(e) a discussion of the extent to which each PRA Element was reviewed, **including a list of SRs that were reviewed**
Section 1-7			
References	See global comment on references at start of Table A-1.		
Appendix 1-A			
Global	The word "significant" is used in many places throughout the Appendix. For example, the term "significant changes in scope or capability" is used to classify a change as a PRA upgrade, rather than a PRA maintenance. The term "significant change in risk insights" is used to indicate when a focused peer review is suggested even for what is nominally classified as a PRA maintenance. While what is meant by the former is clarified in the examples, what constitutes a "significant change in risk insights" needs to be defined and added to the defined terms in Section 1-2.	Clarification	Add to list of definitions -- ***Significant change in risk insights:*** **Whether a change is considered significant is dependent on the context in which the insights are used. A change in the risk insights is considered significant when it has the potential to change a decision being made using the PRA.**

Table A-1. Staff Position on ASME/ANS RA-Sa-2009 Part 1,
General Requirements for an At-Power Level 1 and LERF PRA

Index No	Issue	Position	Resolution
1-A.3, Examples 1 thru 7, 9, 11-16, 19, 20, 22 thru	--------------------	No objection	--------------------
1-A.1, 4th paragraph	As written, it could be inferred that a newly developed method would not be considered an upgrade.	Clarification	. . . "new" should be interpreted as new to the subject PRA even though the methodology in question has been applied in other PRAs **and includes newly developed methods that have been used in the base PRA by the analyst**. ~~It is not intended to imply a NDM~~. This interpretation . . .
1-A.2	An "internal review" is recommended in several places. This recommendation is made instead of an "outside" peer review. It needs to be made clear that this internal review is a type of "peer review" and **should** follow the process and requirements for the peer review requirements.	Clarification	(d) In the context . . . A focused review would be warranted. **(e) When performing an internal review, the objective is to assess that the change to the PRA was correctly performed. In performing this assessment, the reviewer should use as guidance those applicable requirements in the standard.**
1-A.3, Examples 8, 10, 17	It is assumed that a change to the base PRA that involves a calculation using the same computer code is a PRA maintenance type change rather than a PRA upgrade type change. This assumption would only be valid if the calculation does not involve any new assumptions and the same analyst is performing the calculation.	Clarification	*Change:* . . . . using the same computer code that was used for the prior calculations**, given the calculation does not involve any new assumptions and the calculation is performed using the same guidance.** **NOTE: the words "that was used for the prior calculations" do not appear in Example #8, staff clarification includes these words in Example #8.**

Table A-1. Staff Position on ASME/ANS RA-Sa-2009 Part 1, General Requirements for an At-Power Level 1 and LERF PRA

Index No	Issue	Position	Resolution
1-A.3, Example 18	Changing the definition of core damage without changing the thermal-hydraulic methodology may result in changed success criteria which could change the accident progression delineated by the accident sequences. It is not a foregone conclusion that this is a simple change to the PRA model. It needs to be reviewed to ensure that the resulting changes are appropriate. Further, what would be a significant change is open to interpretation, and "would be prudent" is not as strong as "should."	Clarification	*Discussion and/or Alternative Recommendation:* **While this change may not be a "new methodology," it could result in changing the success criteria with implications for the development of accident sequences, and potentially on the HRA (through timing), data, and quantification.** If this change leads to a significant change in risk insights, a focused peer review should be performed
1-A.3, Example 21	This assumes that the "important" human actions are of the same nature as the new ones being added and utilize the ASEP method in the exact same manner. This cannot be assumed.	Clarification	*Rationale:* **If it can be shown that the previous "important" human actions fully utilized the ASEP method, and that any deficiencies by the analyst were corrected, then, if** there is no significant impact on risk insights, this change falls into ....
1-A.4	References	Clarification	See global comment on references at start of Table A-1.

Table A-2. Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events

Index No	Issue	Position	Resolution
Section 2-1			
2-1.1 thru 2-1.3	--------------------	No objection	--------------------
Section 2-2			
2-2.1	--------------------	No objection	--------------------
2-2.1 – IE			
2-2.1.1	--------------------	No objection	--------------------
Table 2-2.1-1	--------------------	No objection	--------------------
Tables 2-2.1-2(a) thru 2-2.1-5(d)			
IE-A1 thru IE-A4, IE-A7 thru IE-A10	--------------------	No objection	--------------------
IE-A5	The search for initiators should go down to the subsystem/train level. Capability Category III should consider the use of "other systematic processes."	Clarification	<u>Cat I and II</u>: PERFORM a systematic evaluation of each system **and where necessary down to the subsystem or train level**, including support systems…. <u>Cat III</u>: PERFORM a systematic evaluation of each system **down to the subsystem or train level**, including support systems…. PERFORM an FMEA (failure modes and effects analysis) **or other systematic process** to assess….

Table A-2. Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events

Index No	Issue	Position	Resolution
IE-A6	Initiating events from common cause or from both routine and non-routine system alignments should be considered.	Clarification	Cat II: …resulting from multiple failures~~,~~ if the equipment failures result from a common cause, ~~and~~ **or** from routine system alignments **resulting from preventive and corrective maintenance**. Cat III: …resulting from multiple failures, including equipment failures resulting from random and common causes, ~~and~~ **or** from ~~routine~~ system alignments **resulting from preventive and corrective maintenance**.
IE-B1 thru IE-B5	--------------------	No objection	--------------------
IE-C1 thru IE-C11, IE-C13 thru IE-C15	--------------------	No objection	--------------------
IE-C12	Providing a list of generic data sources would be consistent with other SRs related to data.	Clarification	COMPARE results and EXPLAIN differences in the initiating event analysis with generic data sources to provide a reasonable check of the results. **An example of an acceptable generic data sources is NUREG/CR-6928 [Note (1)].**
Footnote (1)(a) to Table 2-2.1-4(c)	The first example makes an assumption that the hourly failure rate is applicable for all operating conditions.	Clarification	…Thus, $f_{\text{bus at power}} = 1\times10^{-7}$/hr * 8760 hrs/yr * 0.90 = 7.9×10^{-4}/reactor year. **In the above example, it is assumed the bus failure rate is applicable for at-power conditions. It should be noted that initiating event frequencies may be variable from one operating state to another due to various factors. In such cases, the contribution from events occurring only during at-power conditions should be utilized.**
IE-D1 thru IE-D3	--------------------	No objection	--------------------

Table A-2. Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events

Index No	Issue	Position	Resolution
2-2.2 – AS			
2-2.2.1	The HLR and associated SRs are written for CDF and not LERF; therefore, references to LERF are not appropriate.	Clarification	2-2.2.1 Objectives. The objectives… reflected in the assessment of CDF ~~and LERF~~ is such a way that….
Table 2-2.2-1	--------------------	No objection	--------------------
Tables 2-2.2-2(a) thru 2-2.2-4(c)			
AS-A1 thru AS-A8, AS-A10, AS-A11	--------------------	No objection	--------------------
AS-A9	The code requirements for acceptability need to be stated.	Clarification	<u>Cat II and III</u>: …affect the operability of the mitigating systems. **(See SC-B4.)**
AS-B1 thru AS-B7	--------------------	No objection	--------------------
AS-C1 thru AS-C3	--------------------	No objection	--------------------
2-2.3 – SC			
2-2.3.1	The HLR and associated SRs are written for CDF and not LERF; therefore, references to LERF are not appropriate.	Clarification	(a) overall success criteria are defined (i.e., core damage ~~and large early release~~)
Table 2-2.3-1	--------------------	No objection	--------------------
Tables 2-2.3-2(a) thru 2-2.3-4(c)			
SC-A1 thru SC-A6	-------------------- Note: SC-A3 was deleted in Addendum B.	No objection	--------------------
SC-B1 thru SC-B5	--------------------	No objection	--------------------
SC-C1 thru SC-C3	--------------------	No objection	--------------------

Table A-2. Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events

Index No	Issue	Position	Resolution
2-2.4 – SY			
2-2.4.1	--------------------	No objection	--------------------
Table 2-2.4-1	--------------------	No objection	--------------------
Tables 2-2.4-2(a) thru 2-2.4-4(c)			
SY-A1 thru SY-A23	--------------------	No objection	--------------------
SY-A24	There are no commonly used analysis methods for recovery in the sense of repair, other than use of actuarial data.	Clarification	…is justified through an adequate analysis or examination of data **collected in accordance with DA-C15 and estimated in accordance with DA-D9.** ~~(See DA-C15.)~~
SY-B1 thru SY-B13, SY-B15	--------------------	No objection	--------------------
SY-B14	Containment vent and failure can cause more than NPSH problems (e.g., harsh environments).	Clarification	Examples of degraded environments include: **(h) harsh environments induced by containment venting, failure of the containment venting ducts, or failure of the containment boundary that may occur prior to the onset of core damage**
SY-C1 thru SY-C3	--------------------	No objection	--------------------
2-2.5 – HR			
2-2.5.1	--------------------	No objection	--------------------
Table 2-2.5-1	--------------------	No objection	--------------------
Tables 2-2.5-2(a) thru 2-2.5-10(i)			
HR-A1 thru HR-A3	--------------------	No objection	--------------------
HR-B1, HR-B2	--------------------	No objection	--------------------
HR-C1 thru HR-C3	--------------------	No objection	--------------------

Table A-2. Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events

Index No	Issue	Position	Resolution
HR-D1, HR-D2HR-D4, HR-D5, HR-D7	--------------------	No objection	--------------------
HR-D3	Add examples for what is meant by quality in items (a) and (b) of Cat II, III.	Clarification	<u>Cat II, III</u>: (a) the quality **(e.g., format, logical structure, ease of use, clarity, and comprehensiveness)** of written procedures (for performing tasks) **and the type of administrative controls that support independent review (e.g., configuration control process, technical review process, training processes, and management emphasis on adherence to procedures).** ~~of administrative controls (for independent review)~~ (b) the quality of the human-machine interface **(e.g., adherence to human factors guidelines [Note (3)] and results of any quantitative evaluations of performance per functional requirements)**, including both the equipment configuration, and instrumentation and control layout **(3) NUREG-0700, Rev. 2, Human-System Interface Design Review Guidelines; J.M. O'Hara, W.S. Brown, P.M. Lewis, and J.J. Persensky, May 2002.**
HR-D6	This SR should be written similarly to HR-G9	Clarification	~~PROVIDE an assessment of the uncertainty in the …. point estimates of HEPs~~. **CHARACTERIZE the uncertainty in the estimates of the HEPs consistent with the quantification approach, and PROVIDE mean values for use in the quantification of the PRA results.**
HR-E1 thru HR-E4	--------------------	No objection	--------------------
HR-F1, HR-F2	--------------------	No objection	--------------------

Table A-2. Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events

Index No	Issue	Position	Resolution
HR-G1, HR-G2, HR-G5 thru HR-G7	--------------------	No objection	--------------------
HR-G3	In item (d) of CC II, III, clarify that “clarity” refers the meaning of the cues, etc. In item (a) of CC I and item (g) of CC II, III, clarify that complexity refers to both determining the need for and executing the required response.	Clarification	Cat I: … (a) the complexity of **detection, diagnosis, decision-making and executing** the **required** response (b) … Cat II, and III: (d) degree of clarity of the cues/indications **in supporting the detection, diagnosis, and decision-making give the plant-specific and scenario-specific context of the event.** (g) complexity of **detection, diagnosis and decision-making, and executing** the required response.
HR-G4	Requirements concerning the use of thermal/hydraulic codes should be cross-referenced.	Clarification	Cat I, II, and III: BASE…. **(See SC-B4.)** SPECIFY the point in time….
HR-G8	Action verb should be capitalized	Clarification	**CHARACTERIZE** ~~Characterize~~ the uncertainty …..
HR-H1 thru HR-H3	--------------------	No objection	--------------------
HR-I1 thru HR-I3	--------------------	No objection	--------------------
2-2.6 - DA			
2-2.6.1	--------------------	No objection	--------------------
Table 2-2.6-1	--------------------	No objection	--------------------
Tables 2-2.6-2(a) thru 2-2.6-6(e)			
DA-A1 thru DA-A4	--------------------	No objection	--------------------

Table A-2. Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events

Index No	Issue	Position	Resolution
DA-B1, DA-B2	--------------------	No objection	--------------------
DA-C1 thru DA-C14, DA-C16	--------------------	No objection	--------------------
DA-C15	This SR provides a justification for crediting equipment repair (SY-A24). As written, it could be interpreted as allowing plant-specific data to be discounted in favor of industry data. In reality, for such components as pumps, plant-specific data is likely to be insufficient and a broader base is necessary.	Qualification	…IDENTIFY instances of plant-specific **experience** ~~or~~ **and, when that is insufficient to estimate failure to repair consistent with DA-D9,** applicable industry experience and for each repair, COLLECT….
DA-D2 thru DA-D8	--------------------	No objection	--------------------
DA-D1	Other approved statistical processes for combining plant-specific and generic data are not available.	Clarification	CC II and III: …USE a Bayes update process ~~or equivalent statistical process that assigns that assigns appropriate weight to the statistical significance of the generic and plant specific evidence and provides an appropriate characterization of the uncertainty.~~ CHOOSE….
DA-D9	New requirement needed, DA-C15 was incomplete, only provided for data collection, not quantification of repair. (See SY-A24.)	Qualification	Cat I, II, and III: **For each SSC for which repair is to be modeled, ESTIMATE, based on the data collected in DA-C15, the probability of failure to repair the SSC in time to prevent core damage as a function of the accident sequence in which the SSC failure appears.**
DA-E1 thru DA-E3	--------------------	No objection	--------------------

Table A-2. Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events

Index No	Issue	Position	Resolution
2-2.7 - QU			
2-2.7.1	SRs for LERF quantification reference the SRs in 2-2.8, and therefore, need to be acknowledged in 2-2.8.	Clarification	The objectives of the quantification element are to provide an estimate of CDF **(and support the quantification of LERF)** based upon the plant-specific… (b) significant contributors to CDF **(and LERF)** are identified such as initiating events…
Table 2-2.7-1 HLR-QU-A, HLR-QU-B, HLR-QU-C, HLR-QU-E, HLR-QU-F	--------------------	No objection	--------------------
Table 2-2.7-1 HLR-QU-D	SRs for LERF quantification reference the SRs in 2-2.8 and, therefore, need to be acknowledged in 2-2.8.	Clarification	…significant contributors to CDF **(and LERF)**, such as initiating events, accident sequences…
Tables 2-2.7-2(a) thru 2-2.7-7(f)			
QU-A1, QU-A4, QU-A5	--------------------	No objection	--------------------
QU-A2	Need to acknowledge LERF quantification	Clarification	…consistent with the estimation of total CDF **(and LERF)** to identify significant accident…
QU-A3	The state-of-knowledge correlation should be accounted for all event probabilities. Left to the analyst to determine the extent of the events to be correlated. Need to also acknowledge LERF quantification	Clarification	Cat I: ESTIMATE the point estimate CDF **(and LERF)** Cat II: ESTIMATE the mean CDF **(and LERF),** accounting for the "state-of-knowledge" correlation between event probabilities ~~when significant~~ (see NOTE 1). Cat III: CALCULATE the mean CDF **(and LERF)** by …

Table A-2. Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events

Index No	Issue	Position	Resolution
QU-B1 thru, , QU-B5, QU-B7 thru QU-B10	--------------------	No objection	--------------------
QU-B6	Need to acknowledge LERF quantification	Clarification	ACCOUNT for … realistic estimation of CDF **or LERF.** This accounting …
QU-C1 thru QU-C3	--------------------	No objection	--------------------
Table 2-2.7-5(d)	HLR-QU-D and Table 2-2.7-2(d) objective statement just before table need to agree; SRs for LERF quantification reference the SRs in 2-2.7 and, therefore, need to be acknowledged in 2-2.7.	Clarification	…significant contributors to CDF **(and LERF)**, such as initiating events, accident sequences…
QU-D1 thru QU-D7	--------------------	No objection	--------------------
QU-E1, QU-E2	--------------------	No objection	--------------------
QU-E3	Need to acknowledge LERF quantification	Clarification	Cat I and II: ESTIMATE the uncertainty interval of the CDF **(and LERF)** results.
QU-E4	The note has no relevance to the base model and could cause confusion; it should be deleted.	Clarification	For each source of model uncertainty … introduction of a new initiating event) ~~[Note (1)].~~ ~~NOTE: For specific applications, … And in logical combinations.~~
QU-F1, QU-F3 thru QU-F6	--------------------	No objection	--------------------

Table A-2. Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events

Index No	Issue	Position	Resolution
QU-F2	SR needs to use defined term "significant" instead of "dominant." In addition, there is no requirement to perform sensitivity studies, and therefore, requirement is not needed for documentation.	Clarification	(g) equipment or human actions that are the key factors in causing the accident~~s~~ **sequences** to be ~~non-dominant~~ **nonsignificant.** ~~(h) the results of all sensitivity studies~~
2-2.8 – LE			
2-2.8.1	--------------------	No objection	--------------------
Table 2-2.8-1	--------------------	No objection	--------------------
Tables 2-2.8-2(a) thru 2-2.8-8(g)			
LE-A1 thru LE-A5	--------------------	No objection	--------------------
LE-B1 thru LE-B3	--------------------	No objection	--------------------
LE-C1 thru LE-C13	--------------------	No objection	--------------------
LE-D1 thru LE-D7	--------------------	No objection	--------------------
LE-E1 thru LE-E4	--------------------	No objection	--------------------
LE-F1 thru LE-F3	--------------------	No objection	--------------------
LE-G1, LE-G3 thru LE-G6	--------------------	No objection	--------------------
LE-G2	There is no requirement to perform sensitivity studies.	Clarification	(h) the model integration … quantification ~~including uncertainty and sensitivity analyses, as appropriate for the level of analysis~~
Table 2-2.8-9	--------------------	No objection	--------------------

Table A-2. Staff Position on ASME/ANS RA-Sa-2009 Part 2, Technical and Peer Review Requirements for At-Power Internal Events

Index No	Issue	Position	Resolution
Section 2-3			
2-3.1 thru 2-3.3.8.2	--------------------	No objection	--------------------
Section 2-4			
References	--------------------	Clarification	See global comment on references at start of Table A-1.

Table A-3. Staff Position on ASME/ANS RA-Sa-2008 Part 3, Technical and Peer Review Requirements for At-Power Internal Flood

Index No	Issue	Position	Resolution
Section 3-1			
3-1.1 thru 3-1.3	--------------------	No objection	--------------------
Section 3-2			
3-2	--------------------	No objection	--------------------
3-2.1 – IFPP			
3-2.1.1	--------------------	No objection	--------------------
Table 3-2.1-1	--------------------	No objection	--------------------
Tables 3-2.1-2(a) thru 3-2.1-3(b)			
IFPP-A1 thru IFPP-A5	--------------------	No objection	--------------------
IFPP-B1 thru IFPP-B3	--------------------	No objection	--------------------
3-2.2 – IFSO			
3-2.2.1	--------------------	No objection	--------------------
Table 3-2.2-1	--------------------	No objection	--------------------
Tables 3-2.2-2(a) thru 3-2.2-3(b)			
IFSO-A2 thru IFSO-A4, IFSO-A6	--------------------	No objection	--------------------
IFSO-A1	The list of fluid systems should be expanded to include fire protection systems.	Clarification	For each flood area ... INCLUDE: (a) equipment (e.g., piping, valves, pumps) located in the area that are connected to fluid systems (e.g., circulating water system, service water system, …~~and~~ reactor coolant system, **and fire protection system**) …

Table A-3. Staff Position on ASME/ANS RA-Sa-2008 Part 3, Technical and Peer Review Requirements for At-Power Internal Flood

Index No	Issue	Position	Resolution
IFSO-A5	It is necessary to consider a range of flow rates for identified flooding sources, each having a unique frequency of occurrence. For example, small leaks that only cause spray are more likely than large leaks that may cause equipment submergence.	Clarification	(b) **range of** flow rate**s**
IFSO-B1 thru IFSO-B3	--------------------	No objection	--------------------
3-2.3 – IFSN			
3-2.3.1	--------------------	No objection	--------------------
Table 3-2.3-1	--------------------	No objection	--------------------
Tables 3-2.3-2(a) thru 3-2.3-3(b)			
IFSN-A1 thru IFSN-A5, IFSN-A-7 thru IFSN-A17	--------------------	No objection	--------------------

Table A-3. Staff Position on ASME/ANS RA-Sa-2008 Part 3, Technical and Peer Review Requirements for At-Power Internal Flood

Index No	Issue	Position	Resolution
IFSN-A6	For Cat II, it is not acceptable to just note that a flood-induced failure mechanism is not included in the scope of the internal flooding analysis. Some level of assessment is required.	Qualification	Cat I: For the SSCs identified in IFSN-A5, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process. EITHER: (a) ASSESS… by using conservative assumptions; OR (b) NOTE that these mechanisms are not included in the scope of the evaluation. **Cat II:** **For the SSCs identified in IFSN-A5, IDENTIFY the susceptibility of each SSC in a flood area to flood-induced failure mechanisms. INCLUDE failure by submergence and spray in the identification process.** **ASSESS qualitatively the impact of flood-induced mechanisms that are not formally addressed (e.g., using the mechanisms listed under Capability Category III of this requirement), by using conservative assumptions.**
IFSN-B1 thru IFSN-B3	--------------------	No objection	--------------------
3-2.4 – IFEV			
3-2.4.1	--------------------	No objection	--------------------
Table 3-2.4-1	--------------------	No objection	--------------------
Tables 3-2.4-2(a) thru 3-2.4-3(b)			
IFEV-A1 thru IFEV-A8	--------------------	No objection	--------------------
IFEV-B1 thru IFEV-B3	--------------------	No objection	--------------------
3-2.5 – IFQU			

Table A-3. Staff Position on ASME/ANS RA-Sa-2008 Part 3, Technical and Peer Review Requirements for At-Power Internal Flood

Index No	Issue	Position	Resolution
3-2.5.1	--------------------	No objection	--------------------
Table 3-2.5-1	--------------------	No objection	--------------------
Tables 3-2.5-2(a) thru 3-2.5-3(b)			
IFQU-A1 thru IFQU-A7, IFQU-A9 thru IFQU-A11	--------------------	No objection	--------------------
IFQU-A8	The quantification also needs to include the effect of common-cause failure.	Clarification	INCLUDE, in the quantification, the combined effects of … including equipment failures, unavailability due to maintenance, **common-cause failures** and other credible causes.
IFQU-B1 thru IFQU-B3	--------------------	No objection	--------------------
Section 3-3			
3-3.1 thru 3-3.3	--------------------	No objection	--------------------
Section 3-4			
References	--------------------	Clarification	See global comment on references at start of Table A-1.

Table A-4. Staff Position on ASME/ANS RA-Sa-2009 Part 4, Technical and Peer Review Requirements for At-Power Internal Fire

Index No	Issue	Position	Resolution
Section 4-1			
4-1.1 thru 4-1.6	--------------------	No objection	--------------------
1.1.1.1.1.1.1 Section 4-2			
4-2	--------------------	No objection	--------------------
4-2.1 – PP			
4-2.1.1, 4-2.1.2	--------------------	No objection	--------------------
Table 4-2.1-1	--------------------	No objection	--------------------
Tables 4-2.1-2(a) thru 4-2.1-4(c)			
PP-A1	--------------------	No objection	--------------------
PP-B1 thru PP-B7	--------------------	No objection	--------------------
PP-C1 thru PP-C4	--------------------	No objection	--------------------
4-2.2 – ES			
4-2.2	--------------------	No objection	--------------------
Table 4-2.2-1 HLR-ES-A	Grammatical change for clarity	Clarification	...identify equipment whose failure, **including spurious operation,** caused by an initiating fire, ~~including spurious operation will~~ **would** contribute …
Tables 4-2.2-2(a) thru 4-2.2.5(d)			
Table 4-2.2-2(a) HLR-ES-A	Conforming change to HLR-ES-A	Clarification	…identify equipment whose failure, **including spurious operation,** caused by an initiating fire, ~~including spurious operation will~~ **would** contribute …
ES-A2 thru ES-A6	--------------------	No objection	--------------------

Table A-4. Staff Position on ASME/ANS RA-Sa-2009 Part 4, Technical and Peer Review Requirements for At-Power Internal Fire

Index No	Issue	Position	Resolution
ES-A1	Conforming change to HLR-ES-A	Clarification	IDENTIFY equipment whose failure, **including spurious operation,** caused by an initiating fire, ~~including spurious operation~~ would contribute …
ES-B2, ES-B3, ES-B5	--------------------	No objection	--------------------
ES-B1	The notes states this requirement is a starting point for selection of mitigating equipment, and that an iterative process will provide the completeness with respect to Table 1-1.3-1, which specifies that the significant contributors be included in the model. The requirement should represent the end result, not the beginning point. Although the definition of failure mode in Part 1 includes spurious operation, it is worth explicitly including since it is an important issue.	Qualification	Cat II: IDENTIFY Fire … and INCLUDE **fire** risk-significant equipment ~~from the internal events PRA~~. NOTE-ES-B1-7: The gradation across … the Fire PRA (other equipment can be assumed failed in the worst possible failure mode**, including spurious operation**). This will tend …
ES-B4	SR refers to incorrect SR	Clarification	… equipment identification per SRs ES-B1 through ES-**B3**~~B~~4.

Table A-4. Staff Position on ASME/ANS RA-Sa-2009 Part 4, Technical and Peer Review Requirements for At-Power Internal Fire

Index No	Issue	Position	Resolution
ES-C1	There is a concern with the way in which the term "significant' has been used. It is ambiguous as to whether the reference is to the total CDF, the internal events CDF, or the fire CDF. In order to avoid ambiguity, it is necessary to have a definition of the term "significant." The terms "significant accident sequence," "significant accident progression sequence," "significant basic event," "significant cutset," and "significant contributor" are defined in Part 1 within the context of the hazard group, so that in Part 3, they should be interpreted as being measured with respect to the fire risk.	Clarification	NOTE-ES-C1-3: … is not a significant contributor **(as defined in Part 1),** …
ES-C2	--------------------	No objection	--------------------
ES-D1	--------------------	No objection	--------------------
4-2.3 – CS			
4-2.3	--------------------	No objection	--------------------
Table 4-2.3-1	--------------------	No objection	--------------------
Tables 4-2.3-2(a) thru 4-2.3-4(c)			
CS-A1 thru CS-A9, CS-A11	--------------------	No objection	--------------------

Table A-4. Staff Position on ASME/ANS RA-Sa-2009 Part 4, Technical and Peer Review Requirements for At-Power Internal Fire

Index No	**Issue**	**Position**	**Resolution**
CS-A10	PP-B1 already allows physical analysis units to be defined in terms of fire areas. As such the distinction between CCI and CCII is unnecessary.	Clarification	~~Cat I:~~ ~~IDENTIFY the fire areas … and CONFIRM … terminal end locations.~~ ~~Cat II:~~ ~~IDENTIFY … and CONFIRM … terminal end locations.~~ **Cat I and II:** **IDENTIFY the physical analysis units, consistent with the plant partitioning analysis, through which each cable associated with a credited Fire PRA function passes *and* CONFIRM that the information includes treatment of cable terminal end locations.**
CS-B1	--------------------	No objection	--------------------
CS-C1 thru CS-C4	--------------------	No objection	--------------------
4-2.4 – QLS			
4-2.4	--------------------	No objection	--------------------
Table 4-2.4-1	--------------------	No objection	--------------------
Tables 4-2.4-2(a) thru 4-2.4-3(b)			
QLS-A1 thru QLS-A4	--------------------	No objection	--------------------
QLS-B1 thru QLS-B3	--------------------	No objection	--------------------
4-2.5 – PRM			
4-2.5	--------------------	No objection	--------------------
Table 4-2.5-1	--------------------	No objection	--------------------
Tables 4-2.5-2(a) thru 4-2.5-4(c)			
PRM-A1 thru PRM-A4	--------------------	No objection	--------------------

Table A-4. Staff Position on ASME/ANS RA-Sa-2009 Part 4, Technical and Peer Review Requirements for At-Power Internal Fire

Index No	Issue	Position	Resolution
PRM-B1 thru PRM-B15	--------------------	No objection	--------------------
PRM-C1	--------------------	No objection	--------------------
4-2.6 – FSS			
4-2.6	--------------------	No objection	--------------------
Table 4-2.6-1	--------------------	No objection	--------------------
Tables 4-2.6-2(a) thru 4-2.6-9(h)			
FSS-A1, FSS-A3, FSS-A6	--------------------	No objection	--------------------
FSS-A2	Need to clarify that spurious operation is a failure mode.	Clarification	…For each target set, SPECIFY …including specification of the failure modes**, including spurious operation**.
FSS-A4	Use of language, "one or more," is problematic, since it does not specify a minimum requirement.	Clarification	IDENTIFY **sufficient** ~~one or more~~ combinations of target sets ... has been represented.
FSS-A5	The number of individual fire scenarios and level of detail should be commensurate with the relative risk importance of the physical analysis unit.	Clarification	<u>Cat I and II</u>: For each unscreened … can be characterized **commensurate with its risk significance**. NOTE FSS-A5-5: It is expected … will be commensurate with ~~the capability category and~~ the fire relative risk importance …
FSS-B1, B2	--------------------	No objection	--------------------
FSS-C1, FSS-C3 thru FSS-C8	--------------------	No objection	--------------------

Table A-4. Staff Position on ASME/ANS RA-Sa-2009 Part 4, Technical and Peer Review Requirements for At-Power Internal Fire

Index No	Issue	Position	Resolution
FSS-C2	See Issue for ES-C1	Clarification	<u>Cat II and III</u>: For those scenarios that represent significant contributors to ~~a physical analysis unit's~~ fire risk, CHARACTERIZE … NOTE FSS-C3-3: … are not significant contributors **(as defined in Part 1),** …
FSS-D1, FSS-D2, FSS-D4 thru FSS-D11	--------------------	No objection	--------------------
FSS-D3	Again the "either bounded or accurately characterized" issue for CC II and CC III.	Clarification	<u>Cat I</u>: …in the analysis of each fire scenario **such that the fire risk contribution of each unscreened physical analysis unit is bounded**. <u>Cat II</u>: …the fire risk contribution of each unscreened physical analysis unit can be ~~either bounded or~~ accurately characterized. <u>Cat III</u>: …the fire risk contribution of each unscreened physical analysis unit can be ~~either bounded or~~ accurately characterized and such that the risk…
FSS-E1 thru FSS-E4	--------------------	No objection	--------------------
FSS-F1	Use of the term "SELECT one or more"	Clarification	<u>Cat II and II</u>: …SELECT ~~one or more fire scenarios(s)~~ **a sufficient number of fire scenarios to characterize** ~~could~~ damage, including collapse, **of** the exposed structural steel…
FSS-F2, FSS-F3	--------------------	No objection	--------------------
FSS-G1 thru FSS-G6	--------------------	No objection	--------------------

Table A-4. Staff Position on ASME/ANS RA-Sa-2009 Part 4, Technical and Peer Review Requirements for At-Power Internal Fire

Index No	Issue	Position	Resolution
FSS-H1 thru FSS-H10	--------------------	No objection	--------------------
4-2.7 – IGN			
4-2.7	--------------------	No objection	--------------------
Table 4-2.7-1	--------------------	No objection	--------------------
Tables 4-2.7-2(a) thru 4-2.7-3(b)			
IGN-A1	The note, IGN-A1-1, appears to be more relevant to IGN-A2 than it is for IGN-A1. Item (e) only makes sense when there is equivalent nuclear experience.	Clarification	NOTE IGN-A1-1…(e) **if being used as a supplement to, rather than in lieu of, nuclear data,** that the fire frequencies calculated are consistent with those derived from nuclear experience ; …
IGN-A2 thru IGN-A10	--------------------	No objection	--------------------
IGN-B1 thru IGN-B5	--------------------	No objection	--------------------
4-2.8 – QNS			
4-2.8	--------------------	No objection	--------------------
Table 4-2.8-1	--------------------	No objection	--------------------
Tables 4-2.8-2(a) thru 4-2.8-5(d)			
QNS-A1	--------------------	No objection	--------------------
QNS-B1, QNS-B2	--------------------	No objection	--------------------

Table A-4. Staff Position on ASME/ANS RA-Sa-2009 Part 4, Technical and Peer Review Requirements for At-Power Internal Fire

Index No	Issue	Position	Resolution
QNS-C1	The screening criteria in Capability Categories II and III should relate to the total CDF and LERF for the fire risk, not the internal events risk. See Issue for 4-2.2-2(c). NOTE ES-C1	Clarification	Cat II: …and • the sum of the CDF contribution for all screened fire compartments is <10% of the estimated total CDF for ~~internal~~ **fire** events and • the sum of the LERF contributions for all screened fire compartments is <10% of the estimated total LERF for ~~internal~~ **fire** events Cat III: …and • the sum of the CDF contributions for all screened fire compartments is <1% of the estimated total CDF for ~~internal~~ **fire** events and • the sum of the LERF contributions for all screened fire compartments is <1% of the estimated total LERF for ~~internal~~ **fire** events
QNS-D1, QNS-D2	--------------------	No objection	--------------------
4-2.9 – CF			
4-2.9	--------------------	No objection	--------------------
Table 4-2.9-1	--------------------	No objection	--------------------
Tables 4-2.9-2(a) thru 4-2.9-5(d)			
CF-A1	See Issue for ES-C1	Clarification	NOTE CF-A1-1: … for non-risk significant contributors **(as defined in Part 1),** …

Table A-4. Staff Position on ASME/ANS RA-Sa-2009 Part 4, Technical and Peer Review Requirements for At-Power Internal Fire

Index No	Issue	Position	Resolution
CF-A2	--------------------	No objection	--------------------
CF-B1	--------------------	No objection	--------------------
4-2.10 – HRA			
4-2.10	--------------------	No objection	--------------------
Table 4-2.10-1	--------------------	No objection	--------------------
Tables 4-2.10-2(a) thru 4-2.10-6(e)			
HRA-A1 thru HRA-A4	--------------------	No objection	--------------------
HRA-B1 thru HRA-B4	--------------------	No objection	--------------------
HRA-C1	--------------------	No objection	--------------------
HRA-D1	--------------------	No objection	--------------------
HRA-D1 [Note (1)]	This SR has the same index number as the previous SR.	Clarification	HRA-D~~1~~**2** [Note (1)]
HRA-E1	--------------------	No objection	--------------------
4-2.11 – SF			
4-2.11	--------------------	No objection	--------------------
Table 4-2.11-1	--------------------	No objection	--------------------
Tables 4-2.11-2(a) thru 4-2.11-3(e)			
SF-A1 thru SF-A5	--------------------	No objection	--------------------
SF-B1	--------------------	No objection	--------------------

Table A-4. Staff Position on ASME/ANS RA-Sa-2009 Part 4, Technical and Peer Review Requirements for At-Power Internal Fire

Index No	Issue	Position	Resolution
4-2.12 – FQ			
4-2.12	--------------------	No objection	--------------------
Table 4-2.12-1 HLR-FQ-E	See Issue for ES-C1	Clarification	HLR-FQ-E: … and significant contributors **(as defined in Part 1)** to CDF and LERF …
Tables 4-2.12-2(a) thru 4-2.12-7(f)			
FQ-A1 thru FQ-A4	--------------------	No objection	--------------------
FQ-B1	--------------------	No objection	--------------------
FQ-C1	--------------------	No objection	--------------------
FQ-D1	--------------------	No objection	--------------------
FQ-E1	See Issue for ES-C1	Clarification	IDENTIFY significant contributors **(as defined in Part 1)** …
FQ-F1	See Issue for ES-C1	Clarification	DOCUMENT the CDF and LERF … • SRs QU-F2 and QU-F3 … are significant contributors **(as defined in Part 1);** …
FQ-F2	--------------------	No objection	--------------------
4-2.13 -- UNC			
4-2.13	--------------------	No objection	--------------------
Table 4-2.13-1	--------------------	No objection	--------------------
UNC-A1, UNC-A2	--------------------	No objection	--------------------

Table A-4. Staff Position on ASME/ANS RA-Sa-2009 Part 4, Technical and Peer Review Requirements for At-Power Internal Fire

Index No	Issue	Position	Resolution
Section 4-3			
4-3.1	--------------------	No objection	--------------------
4-3.2	Expertise in Fire HRA is needed for the peer review	Clarification	…fire modeling, ~~and~~ fire protection programs and their elements**, and Fire HRA**.
4-3.3	--------------------	No objection	--------------------
4-3.3.1 thru 4-3.3.13	--------------------	No objection	--------------------
Section 4-4			
References	--------------------	Clarification	See global comment on references at start of Table A-1.
Appendix 4-A FPRA Methodology (Nonmandatory)			
The staff does not endorse the material in this appendix, and as such, does not have a position (i.e., no objections, no objection with clarification, or no objection with qualification) on any of the material contained in this appendix. However, it should be noted, that consistent with the Commission-endorsed phase PRA Quality Initiative, all risk contributors that cannot be shown as insignificant, should be assessed through quantitative risk assessment methods to support risk informed licensing actions.			

Table A-5. Staff Position on ASME/ANS RA-Sa-2009 Part 5, Technical and Peer Review Requirements for At-Power Seismic Events

Index No	Issue	Position	Resolution
Section 5-1			
5-1	--------------------	No objection	--------------------
Section 5-2			
5-2	--------------------	No objection	--------------------
5-2.1 – SHA			
5-2.1	--------------------	No objection	--------------------
Table 5-2.1.1, HLR-SHA-A thru HLR-SHA-F, HLR-SHA-J	--------------------	No objection	--------------------
Table 5-2.1-1, HLR-SHA-G	Much of the HLR is more how to meet the HLR and should be a SR. Further, the SRs provide the requirements needed in order to meet the HLR. This relationship does not exist here. In addition, this information is also duplicated in the accompanying note. At the least, this text should be removed from the HLR.	Clarification	For further use in the SPRA, the spectral shape SHALL be based on a site-specific evaluation taking into account the contributions of deaggregated magnitude-distance results of the probabilistic seismic hazard analysis. ~~Broad-band, smooth spectral shapes, ... that would challenge these uniform hazard spectral shapes~~.
Table 5-2.1-1, HLR-SHA-H	Much of the HLR is more how to meet the HLR and should be a SR. Further, the SRs provide the requirements needed in order to meet the HLR. This relationship does not exist here.	Clarification	~~When use ... for the intended application.~~ **It shall be confirmed that the basic data and interpretations from an existing study are valid.**
Table 5-2.1-1, HLR-SHA-I	Much of the HLR is more how to meet the HLR and should be a SR. Further, the SRs provide the requirements needed in order to meet the HLR. This	Clarification	~~A screening analysis ... or the magnitude of hazard consequences, or both.~~ **The hazard analysis shall include hazards other than vibratory ground motion if necessary.**

Table A-5. Staff Position on ASME/ANS RA-Sa-2009 Part 5, Technical and Peer Review Requirements for At-Power Seismic Events

Index No	Issue	Position	Resolution
	relationship does not exist here.		
Tables 5-2.1-2(a) to 5-2.1-10(j)			
SHA-A1 thru SHA-A5	--------------------	No objection	--------------------
SHA-B1 thru SHA-B3	--------------------	No objection	--------------------
SHA-C1 thru SHA-C4	--------------------	No objection	--------------------
SHA-D1 thru SHA-D4	--------------------	No objection	--------------------
SHA-E1, SHA-E2	--------------------	No objection	--------------------
SHA-F1 thru SHA-F3	--------------------	No objection	--------------------
Table 5-2.1-8(g)	See issue for Table 5-2.1-1, HLR-SHA-G	Clarification	For further use in the SPRA, the spectral shape SHALL be based on a site-specific evaluation taking into account the contributions of deaggregated magnitude-distance results of the probabilistic seismic hazard analysis. ~~Broad-band, smooth spectral shapes, ... that would challenge these uniform hazard spectral shapes~~.
SHA-G1	Spectral shapes used to evaluate in-structure SSC's must include the effects of amplification from both local site conditions and SSI. Based on IPEEE reviews, certain UHS shapes used for CEUS were not appropriate for the screening purpose.	Clarification	**NOTE HA-G1**: The issue of which spectral shape should be used in the screening of structures, systems, and components (SSCs) and in quantification of SPRA results requires careful consideration. For screening purposes, the spectral shape used should have amplification factors**, including effects from both local site conditions as well as soil-structure interaction,** such that the demand resulting from the use of this shape is higher than that based on the design spectra. This will preclude premature screening of components and will avoid anomalies such as the screened components (e.g., surrogate elements) being the ~~dominant~~ **significant** risk contributing components. Additional discussion on this issue can be found in Ref. 30.

Table A-5. Staff Position on ASME/ANS RA-Sa-2009 Part 5, Technical and Peer Review Requirements for At-Power Seismic Events

Index No	Issue	Position	Resolution
			In the quantification of fragilities and of final risk results, it is important to use as realistic a shape as possible. Semi-site specific shapes, such as those given in NUREG-0098, have been used in the past and ~~are considered~~ **may be** adequate for this purpose**, provided that they are shown to be reasonably appropriate for the site [42]**. The uniform hazard response spectrum (UHS) is acceptable for this purpose **if it can be shown that the UHS shape is appropriate for the site.** ~~unless evidence comes to light (e.g., within the technical literature) that these UHS do not reflect the spectral shape of the site-specific events.~~ **Recent developments [42] indicate that these spectral shapes are not appropriate for CEUS sites where high frequency content is dominant at hard rock sites.**
Table 5-2.1-9(h)	See issue for Table 5-2.1-1, HLR-SHA-H	Clarification	~~When use ... for the intended application.~~ **It shall be confirmed that the basic data and interpretations from an existing study are valid.**
SHA-H	See issue for Table 5-2.1-1, HLR-SHA-H	Clarification	SHA-H**1** <u>Cat I and II</u>: ~~Use of existing studies~~ **ENSURE, in light of established current information, the study meets the requirements in HLR-SHA-A thru HLR-SHA-G**. <u>Cat III</u>: ~~Use of existing studies not allowed~~. **DO NOT USE existing studies.**
Table 5-2.1-10(i)	See issue for Table 5-2.1-1, HLR-SHA-I	Clarification	~~A screening analysis ... or the magnitude of hazard consequences, or both.~~ **The hazard analysis shall include hazards other than vibratory ground motion if necessary.**
SHA-I	See issue for Table 5-2.1-1, HLR-SHA-I	Clarification	~~SHA-I~~

Table A-5. Staff Position on ASME/ANS RA-Sa-2009 Part 5, Technical and Peer Review Requirements for At-Power Seismic Events

Index No	Issue	Position	Resolution
			~~There are no supporting requirements here.~~ **SHA-I1** **Cat I, II and III:** **PERFORM a screening to determine whether to include other seismic hazards such as fault displacement, landslide, soil liquefaction, or soil settlement in the seismic PRA.** *1.1.1.1.1.1.2 SHA-I2* **Cat I, II and III:** **ADDRESS the effect of these other seismic hazards through assessment of the frequency of hazard occurrence or the magnitude of hazard consequences, or both.**
SHA-J1, thru SHA-J3	--------------------	No objection	--------------------
5-2.2 – SFR			
5-2.2	--------------------	No objection	--------------------
5-2.2 Table 5-2.2-1	--------------------	No objection	--------------------
Table 5-2.2-2(a) thru 5-2.2-8(g)			
SFR-A1, SFR-A2	--------------------	No objection	--------------------
SFR-B1, SFR-B2	--------------------	No objection	--------------------
SFR-C1 thru SFR-C6	--------------------	No objection	--------------------
SFR-D1, SFR-D2	--------------------	No objection	--------------------
SFR-E1 thru SFR-E5	--------------------	No objection	--------------------
SFR-F1 thru SFR-F4	--------------------	No objection	--------------------
SFR-G1 thru SFR-G3	--------------------	No objection	--------------------
5-2.3 – SPR			
5-2.3	--------------------	No objection	--------------------

Table A-5. Staff Position on ASME/ANS RA-Sa-2009 Part 5, Technical and Peer Review Requirements for At-Power Seismic Events

Index No	Issue	Position	Resolution
5-2.3 Table 5-2.3-1	--------------------	No objection	--------------------
Tables 5-2.3-2(a) thru 5-2.3-7(f)			
SPR-A1 thru SPR-A4	--------------------	No objection	--------------------
SPR-B1 thru SPR-B11	--------------------	No objection	--------------------
SPR-C1	--------------------	No objection	--------------------
SPR-D1	--------------------	No objection	--------------------
SPR-E1 thru SPR-E6	--------------------	No objection	--------------------
SPR-F1 thru SPR-F3	--------------------	No objection	--------------------
Section 5-3			
5-3	--------------------	No objection	--------------------
Section 5-4			
	References	Clarification	See global comment on references at start of Table A-1.
Appendix 5-A			
5-A.1 thru 5-A.3	--------------------	No objection	--------------------
5-A.4	References	Clarification	See global comment on references at start of Table A-1.

Table A-6. Staff Position on ASME/ANS RA-Sa-2009 Part 6, Technical and Peer Review Requirements for At-Power Screening and Conservative Analysis of Other External Hazards

Index No	Issue	Position	Resolution
Section 6-1			
6-1	--------------------	No objection	--------------------
Section 6-2			
6-2.1 thru 6-2.3	--------------------	No objection	--------------------
Table 6-2-1	--------------------	No objection	--------------------
Tables 6-2-2(a) to 6-2-6(e)			
EXT-A1, EXT-A2	--------------------	No objection	--------------------
EXT-B1 thru EXT-B4	--------------------	No objection	--------------------
EXT-C1 thru EXT-C7	--------------------	No objection	--------------------
EXT-D1, EXT-D2	--------------------	No objection	--------------------
EXT-E1, EXT-E2	--------------------	No objection	--------------------
Section 6-3			
6-3.1 thru 6-3.3	--------------------	No objection	--------------------
Section 6-4			
	References	Clarification	See global comment on references at start of Table A-1.
Appendix 6-A			
	--------------------	No objection	--------------------
6-A-1	References	Clarification	See global comment on references at start of Table A-1.

Table A-7. Staff Position on ASME/ANS RA-Sa-2009, Part 7, Technical and Peer Review Requirements for At-Power High Wind Events

Index No	Issue	Position	Resolution
1.1.1.1.1.1.3 Section 7-1			
7-1	--------------------	No objection	--------------------
Section 7-2			
7-2	--------------------	No objection	--------------------
7-2.1 – WHA			
7-2.1	--------------------	No objection	--------------------
Table 7-2.1-1	--------------------	No objection	--------------------
Tables 7-2.1-2(a) and 7-2.1-2(b)			
WHA-A1	The six elements described in NOTE WIND-A1 provide the details required for the tornado wind hazard analysis and should be included in WIND-A1 as requirements.	Qualification	Cat II and III: In the tornado wind hazard analysis, USE … a mean hazard curve can be derived. **INCLUDE the following elements in the tornado wind hazard analysis:** **(1) Variation of tornado intensity with occurrence frequency (The frequency of tornado occurrence decreases rapidly with increased Intensity);** **(2) Correlation of tornado width and length of damage area; longer tornadoes are usually wider;** **(3) Correlation of tornado area and intensity; stronger tornadoes are usually larger than weaker tornadoes;** **(4) Variation in tornado intensity along the damage path length; tornado intensity varies throughout its life cycle;** **(5) Variation of tornado intensity across the tornado path width.** **(6) Variation of tornado differential pressure across the tornado path width.** **NOTE WIND-A1:** State-of-the-art methodologies are given ... can be found in Refs. 13, 56, and 57.

Table A-7. Staff Position on ASME/ANS RA-Sa-2009, Part 7, Technical and Peer Review Requirements for At-Power High Wind Events

Index No	Issue	Position	Resolution
			~~Tornado wind hazard analysis SHOULD include the following elements:~~ ~~(a) variation of tornado intensity with occurrence …~~ ~~(f) variation of tornado differential pressure across the tornado path width.~~
WHA-A2 thru WHA-A5	--------------------	No objection	--------------------
WHA-B1 thru WHA-B3	--------------------	No objection	--------------------
7-2.2 – WFR			
7-2.2	--------------------	No objection	--------------------
Table 7-2.2-1	--------------------	No objection	--------------------
Tables 7-2.2-2(a thru 7-2.2-3(b))			
WFR-A1, WFR-A2	--------------------	No objection	--------------------
WFR-B1 thru WFR-B3	--------------------	No objection	--------------------
7-2.3 – WPR			
7-2.3	--------------------	No objection	--------------------
Table 7-2.3-1 HLR-WPR-A	The word 'significant' should be added in this HLR in Table 7-2.3 and in the HLR statement in Table 7-2.3-2(a)	Clarification	The wind-PRA systems model shall include wind-caused **significant** initiating events and other failures **that are significant contributors** that can …
Table 7-2.3-1 HLR-WPR-B and HLR-WPR-C	--------------------	No objection	--------------------
Tables 7-2.3-2(a) thru 7-2.3-4(c)			
Table 7-2.3-2(a)	The word 'significant' should be added in the	Clarification	The wind-PRA systems model shall include wind-caused **significant** initiating events and other failures

Table A-7. Staff Position on ASME/ANS RA-Sa-2009, Part 7, Technical and Peer Review Requirements for At-Power High Wind Events

Index No	Issue	Position	Resolution
	HLR statement in Table 7-2.3-2(a)		**that are significant contributors** that can …
WPR-A1thru WPR- A11	--------------------	No objection	--------------------
WPR-B1, WPR- B2	--------------------	No objection	--------------------
WPR-C1 thru WPR- C3	--------------------	No objection	--------------------
Section 7-3			
7-3 thru 7-3.3.5	--------------------	No objection	--------------------
Section 7-4			
	References	Clarification	See global comment on references at start of Table A-1.

Table A-8. Staff Position on ASME/ANS RA-Sa-2009, Part 8, Technical and Peer Review Requirements for At-Power External Flood Events

Index No	Issue	Position	Resolution
Section 8-1			
8-1	--------------------	No objection	--------------------
Section 8-2			
8-2	--------------------	No objection	--------------------
8-2.1 – XFHA			
8-2.1			
Table 8-2.1-1	--------------------	No objection	--------------------
Tables 8-2-2(a) and 8-2.1-3(b)			
Table 8-2-2(a)	Incorrect table number	Clarification	Table ~~8-2-2(a)~~ **8-2.1-2(a)**
XFHA-A1 thru XFHA-A6	--------------------	No objection	--------------------
XFHA-B1 thru XFHA-B3	--------------------	No objection	--------------------
8-2.2 – XFFR			
8-2.2			
Table 8-2.2-1	--------------------	No objection	--------------------
Tables 8-2-2(a) and 8-2.2-3(b)			
Table 8-2-2(a)	Incorrect table number	Clarification	Table ~~8-2-2(a)~~ **8-2.2-2(a)**
XFFR-A1, XFFR-A2	--------------------	No objection	--------------------
XFFR-B1 thru XFFR-B3	--------------------	No objection	--------------------
8-2.3			
8-2.3	--------------------	No objection	--------------------
Table 8-2.3-1 HLR-XFPR-A	The word 'significant' needs to be added in this HLR in Table 8-2.3 and in the HLR statement in Table 8-2.3-2(a)	Clarification	The external flooding-PRA systems model shall include wind-caused **significant** initiating events and other failures **that are significant contributors** that can …
Tables 8-2.3-2(a) and 8-2.3-4(c)			
Table 8-2.3-2(a)	The word 'significant' needs to be added the HLR statement in Table 8-2.3-2(a)	Clarification	The external flooding-PRA systems model shall include wind-caused **significant** initiating events and other failures **that are significant contributors** that can …

Table A-8. Staff Position on ASME/ANS RA-Sa-2009, Part 8, Technical and Peer Review Requirements for At-Power External Flood Events

Index No	Issue	Position	Resolution
XFPR-A thru XFPR-A11	--------------------	No objection	--------------------
XFPR-B1, XFPR-B2	--------------------	No objection	--------------------
XFPR-C1 thru XFPR-C3	--------------------	No objection	--------------------
Section 8-3			
8-3 thru 8-3.3.5	--------------------	No objection	--------------------
Section 8-4			
	References	Clarification	See global comment on references at start of Table A-1.

Table A-9. Staff Position on ASME/ANS RA-Sa-2009, Part 9, Technical and Peer Review Requirements for At-Power Other External Hazards

Index No	Issue	Position	Resolution
1.1.1.1.1.1.4 Section 9-1			
9-1	--------------------	No objection	--------------------
Section 9-2			
9-2	--------------------	No objection	--------------------
9-2.1 – XHA			
9-2.1			
Table 9-2.1-1	--------------------	No objection	--------------------
Tables 9-2.1-2(a) and 9-2.1-3(b)			
XHA-A1 thru XFHA-A4	--------------------	No objection	--------------------
XHA-B1 thru XHA-B3	--------------------	No objection	--------------------
9-2.2 – XFR			
9-2.2			
Table 9-2.2-1	--------------------	No objection	--------------------
Tables 9-2.2-2(a) and 9-2.2-3(b)			
XFR-A1, thru XFFR-A4	--------------------	No objection	--------------------
XFR-B1 thru XFR-B3	--------------------	No objection	--------------------
9-2.3 – XPR			
9-2.3	--------------------	No objection	--------------------
Table 9-2.3-1 HLR-XPR-A	The word 'significant' should be added in this HLR in Table 9-2.3-1 and in the HLR statement in Table 9-2.3-2(a)	Clarification	The external hazard PRA plant model shall include wind-caused **significant** initiating events and other failures **that are significant contributors** that can … shall include wind-caused **significant** initiating events and other failures **that are significant contributors** that can …
Tables 9-2.3-2(a) and 9-2.3-4(c)			
Table 9-2.3-2(a)	The word 'significant' should be added in the HLR statement in Table 9-2.3-2(a)	Clarification	The external hazard PRA plant model shall include wind-caused **significant** initiating events and other failures

Table A-9. Staff Position on ASME/ANS RA-Sa-2009, Part 9, Technical and Peer Review Requirements for At-Power Other External Hazards

Index No	Issue	Position	Resolution
			that are significant contributors that can …
XPR-A thru XPR-A11	--------------------	No objection	--------------------
XPR-B1 thru XFPR-B2	--------------------	No objection	--------------------
XPR-C1 thru XPR-C3	--------------------	No objection	--------------------
Section 9-3			
9-3.1 thru 9-3.4.5	--------------------	No objection	--------------------
Section 9-4			
	References	Clarification	See global comment on references at start of Table A-1.

Table A-10. Staff Position on ASME/ANS RA-Sa-2009 Part 10, Technical and Peer Review Requirements for At-Power Seismic Margins Assessment

The staff does not endorse the material in this Part of the standard, and as such, does not have a position (i.e., no objections, no objection with clarification, or no objection with qualification) on any of the material contained in Part 10 of the standard. However, it should be noted, that consistent with the Commission-endorsed phased PRA Quality Initiative, all risk contributors that cannot be shown as insignificant, should be assessed using a PRA (as defined in regulatory position C.1) to support risk-informed licensing actions.

REFERENCES [12]

1. American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RA-S-2008, ASME, New York, NY, ANS, La Grange Park, Illinois, February 2009.

12 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

APPENDIX B

NRC REGULATORY POSITION ON THE AMERICAN SOCIETY OF MECHANICAL ENGINEERS AND THE AMERICAN NUCLEAR SOCIETY RA-S CASE 1

B-1 Introduction

The American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) have published ASME/ANS RA-S Case 1, "Case for ASME/ANS RA-Sb-2013 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment of Nuclear Power Plant Applications," dated November 22, 2017 (Ref. 1), which is a proposed alternative set of requirements related to requirements for a seismic probabilistic risk assessment (PRA) in Part 5 of ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Ref. 2). The standard states that it "sets forth requirements for probabilistic risk assessments (PRAs) used to support risk-informed decision for commercial nuclear power plants and describes a method for applying these requirements for specific applications." The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed ASME/ANS RA-S Case 1 against the relevant characteristics and attributes for an acceptable PRA as discussed in regulatory positions C.1 and C.2 of this regulatory guide (RG). The staff's position on each requirement (referred to in the standard as a requirement, a high-level requirement, or a supporting requirement) in ASME/ANS RA-S Case 1 is categorized as "no objection," "no objection with clarification," or "no objection subject to the following qualification," and defined as follows:

- **No objection.** The staff has no objection to the requirement.

- **No objection with clarification.** The staff has no objection to the requirement. However, certain requirements, as written, are either unclear or ambiguous, and therefore the staff has provided its understanding of these requirements.

- **No objection subject to the following qualification.** The staff has a technical concern with the requirement and has provided a qualification to resolve the concern.

Table B-1 provides the staff's position on each requirement. A discussion of the staff's concern (issue) and the staff proposed resolution is provided. In the proposed staff resolution, the staff clarification or qualification to the requirement is indicated in either bolded text (i.e., **bold**) or strikeout text (i.e., ~~strikeout~~); that is, the necessary additions or deletions to the requirement (as written in ASME/ANS RA-S Case 1) for the staff to have no objection are provided.

Table B-1. Staff Position on ASME/ANS RA-S Case 1 Technical and Peer Review Requirements of a Level 1, Seismic PRA for the At-Power Operating Mode

Index No	Issue	Position	Resolution
Section 5-1			
5-1.1 to 5-1.2	--------------------	No objection	--------------------
5-1.3	The last paragraph of the section states that the internal events PRA model is the starting point "…to which must be added a number of structures, systems, and components (SSCs) not included in the model but that could fail due to the external hazard." Failure modes caused by the external hazard for SSCs existing in the internal events PRA should also be included.	Clarification	The approach to any external hazard PRA typically uses as its starting point the internal events PRA model "to which must be added a number of structures, systems, and components (SSCs) not included in the model but that could fail due to the external hazard **and new failure modes caused by the external hazard for SSCs already present in the model**." Both the part of the internal events model dealing with CDF and the part dealing with LERF are used as starting points.
Eliminated Sections 5-1.4 and 5-1.5	--------------------	No objection	--------------------
Eliminated Section 5-1.6	The Part 5 Code Case does not include the language from Section 5-1.6 in ASME/ANS RA-Sb-2013, which discussed the usage of generic fragility information. Section 5-1.6 in ASME/ANS RA-Sb-2013 indicates that "(a) Analysts should apply caution in the use of generic fragilities and provide justification that the generic fragilities are applicable, and (b) Peer reviews should focus on the use of generic fragilities to ensure that their use is appropriate and justified." These	Clarification	Include in the nonmandatory appendix (NMA) language on the use of generic fragility information as in Section 5-1.6 in ASME/ANS RA-Sb-2013 as follows: **(a) Analysts should apply caution in the use of generic fragilities and provide justification that the generic fragilities are applicable, and (b) Peer reviews should focus on the use of generic fragilities to ensure that their use is appropriate and justified.**

Table B-1. Staff Position on ASME/ANS RA-S Case 1 Technical and Peer Review Requirements of a Level 1, Seismic PRA for the At-Power Operating Mode

Index No	Issue	Position	Resolution
	statements are important because they appropriately identify the scope of interest with respect to generic fragility for both the analysts and the peer reviewers.		
Section 5-2			
Introductory text	Text was removed from Section 5-2 that helps set the context for the standard requirements.	Clarification	Seismic PRA is an integrated activity requiring close interactions among specialists from different fields (e.g., seismic hazard analysis, systems analysis, fragility evaluation). For this reason, it is important that all members of the seismic PRA team be cognizant of all of the supporting requirements (SRs) in this part, not just those in their area of expertise, and understand the interactions required between the elements. **The analysis requires judgment and extrapolation beyond observed data. Therefore, the analyst is strongly urged to review published seismic PRA reports and to compare his/her plant-specific seismic PRA to the published studies of similar reactor types and system designs.** This understanding of the standard **and other seismic PRAs** will promote consistency among similar PRAs and risk-informed applications and will also promote reasonableness in the numerical results and risk insights. The peer review is also directed in part toward this same objective of reasonableness in the numerical results and risk insights.
Section 5-2.1			
Introductory text	The first full paragraph of Section 5-2.1 states in part, "The requirements described in Part 5-2.1 address these objectives in detail. A probabilistic seismic hazard analysis (PSHA), which may directly incorporate site	Clarification	The requirements described in Part 5-2.1 address these objectives in detail. A probabilistic seismic hazard analysis (PSHA)~~, which may directly incorporate site response analyses,~~ is used to assess horizontal ground motions at the site.

Table B-1. Staff Position on ASME/ANS RA-S Case 1 Technical and Peer Review Requirements of a Level 1, Seismic PRA for the At-Power Operating Mode

Index No	Issue	Position	Resolution
	response analyses, is used to assess horizontal ground motions at the site." It does not seem appropriate to highlight a specific aspect of the PSHA, particularly in such an ambiguous manner.		
General comments on the SHA Technical Element	The Code Case proposes definitions for the terms "primary hazard" and "secondary hazard." However, the Code Case only uses the term "primary hazard" in the definition of the term "secondary hazard," which may not prompt a need to define the term "primary hazard." The primary hazard described by the objectives in Section 5-2.1 seems to be the vibratory ground motion. However, in many instances, but not all, the text refers to secondary hazards from vibratory ground motions. It is unclear whether there is a difference between the way vibratory ground motion is referred to in primary and secondary hazards or if these are intended to be synonymous. Consideration should be given to whether the definition be made more precise to the hazards, primary or secondary, that the Code Case intends to address. For example, does it intend to address tsunamis and seiches? If	Clarification	• Ensure consistent use of the term "secondary hazard" with the definition. • To the extent possible express which secondary seismic hazards are included or, alternatively, which are not.

Table B-1. Staff Position on ASME/ANS RA-S Case 1 Technical and Peer Review Requirements of a Level 1, Seismic PRA for the At-Power Operating Mode

Index No	Issue	Position	Resolution
	not, it should not be mentioned.		
Table 5-2.1-1			
HLR-SHA-A	The language of the high-level requirement (HLR) HLR-SHA-A states, "The frequency of seismic ground motion at the site shall be based on a site-specific PSHA that represents the center, body, and range of the technically defensible interpretations. The level of analysis, as well as the level of updates when an existing study is the initial basis for the site-specific PSHA, shall be determined based on the intended application and on the technical viability of existing PSHA models." This language is too vague. In particular, the frequency of the ground motion is a natural process. It is the frequency of the ground motion calculation that is based on a PSHA.	Clarification	**The basis for the calculation of** ~~T~~the frequencies of **exceeding different levels of vibratory** seismic ground motion at the site shall be ~~based on~~ a site-specific PSHA that represents the center, body, and range of the technically defensible interpretations. The level of analysis, as well as the level of updates when an existing study is the initial basis for the site-specific PSHA, shall be determined based on the intended application and on the technical viability of existing PSHA models.
HLR-SHA-B through HLR-SHA-J	--------------------	No objection	--------------------
Table 5-2.1-2			
Introductory text	--------------------	No objection	--------------------
SHA-A1 through SHA-A4	--------------------	No objection	--------------------
SHA-A5	Regarding supporting requirement SHA-A5 in Table 5-2.1-2, the NRC staff has discouraged use of the damage parameter cumulative absolute	Clarification	JUSTIFY the specified lower-bound magnitude ~~(or probabilistically defined characterization of magnitudes based on a damage parameter)~~ for use in the hazard analysis, such that earthquakes of magnitudes less than this value

Table B-1. Staff Position on ASME/ANS RA-S Case 1 Technical and Peer Review Requirements of a Level 1, Seismic PRA for the At-Power Operating Mode

Index No	Issue	Position	Resolution
	velocity (CAV) filter in place of a lower bound magnitude for the PSHA. Use of CAV has often been misapplied in PSHAs to improperly filter out larger magnitude events at larger source-to-site distances. Recently completed PSHAs for Fukushima Near-Term Task Force (NTTF) Recommendation 2.1 and combined operating license (COL) and early site permit (ESP) applications no longer use the CAV damage parameter in place of a lower bound magnitude. The NRC staff's related letter pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.54(f) specified use of *M5* (moment magnitude 5) as an appropriate lower bound magnitude.		are not expected to cause significant damage to the engineered structures or equipment.
SHA-A6	--------------------	No objection	--------------------
Note (1), Issue 1	Note (1) of Table 5-2.1-2 states, in part, "The appropriate level of the hazard analysis will depend on project-specific factors and should include considerations such as the safety significance of the nuclear power plant, the technical complexity and uncertainties in hazard inputs, regulatory oversight and requirements, and the availability of resources." Although it is a note and	Clarification	The appropriate level of the hazard analysis will depend on project-specific factors and should include considerations such as the ~~safety significance of the nuclear power plant~~ **seismicity of the plant's location, the plant's seismic ruggedness**, **and** the technical complexity and uncertainties in hazard inputs~~,~~ ~~regulatory oversight and requirements, and the availability of resources~~**.**

Table B-1. Staff Position on ASME/ANS RA-S Case 1 Technical and Peer Review Requirements of a Level 1, Seismic PRA for the At-Power Operating Mode

Index No	Issue	Position	Resolution
	not a requirement, citing the availability of resources as a means of determining the appropriate level of hazard analysis may be misconstrued as a justification for excluding consideration of a safety issue.		
Note (1), Issue 2	Note (1) of Table 5-2.1-2 refers to RG 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," as providing an acceptable approach to establishing a lower-bound magnitude for use in the hazard analysis. However, as discussed above with regard to SHA-A5, the NRC staff has discouraged use of the damage parameter CAV filter in place of a lower bound magnitude for the PSHA. Use of CAV has often been misapplied in PSHAs to improperly filter out larger magnitude events at larger source-to-site distances. Recently completed PSHAs for NTTF Recommendation 2.1 and COL and ESP applications no longer use the CAV damage parameter in place of a lower bound magnitude. The NRC staff's related letter pursuant to 10 CFR 50.54(f) specified use of *M*5 (moment magnitude 5) as an	Clarification	Remove the following language in Note (1) of Table 5-2.1-2: ~~RG 1.208 [5-3] provides one acceptable approach to establishing a lower-bound magnitude for use in the hazard analysis.~~

Table B-1. Staff Position on ASME/ANS RA-S Case 1 Technical and Peer Review Requirements of a Level 1, Seismic PRA for the At-Power Operating Mode

Index No	Issue	Position	Resolution
	appropriate lower-bound magnitude.		
Table 5-2.1-3			
Introductory text	--------------------	No objection	--------------------
SHA-B1, SHA-B2	--------------------	No objection	--------------------
SHA-B3	Sole use of term “attenuation” in conjunction with modeling ground motions is unnecessarily limiting.	Clarification	ENSURE that the data and information are sufficient to characterize attributes important for modeling both regional **propagation** ~~attenuation~~ of ground motions and local site effects including their associated uncertainties.
SHA-B4	--------------------	No objection	--------------------
SHA-B5	The current language requires a demonstration that the updated earthquake catalog has been reviewed if an existing PSHA is used. However, this does not include accounting for the impact of the updated earthquake catalog on the existing PSHA.	Clarification	If an existing PSHA is used, DEMONSTRATE that an updated catalog of earthquakes ~~was reviewed in the evaluation to determine if~~ **does not make** the existing PSHA ~~remains~~ **un**viable.
Notes	--------------------	No objection	--------------------
Table 5-2.1-4			
Introductory text	--------------------	No objection	--------------------
SHA-C1 through SHA-C4	--------------------	No objection	--------------------
Notes	--------------------	No objection	--------------------
Table 5-2.1-5			
Introductory text	--------------------	No objection	--------------------
SHA-D1	The ground motion characterization model needs to include the interface with the site response analysis in terms of a reference soil or rock	Clarification	In the ground motion characterization model that determines the range of seismic vibratory ground motion that can occur at a site, INCLUDE

Table B-1. Staff Position on ASME/ANS RA-S Case 1 Technical and Peer Review Requirements of a Level 1, Seismic PRA for the At-Power Operating Mode

Index No	Issue	Position	Resolution
	horizon, as defined by shear wave velocity, density, and damping values.		*(a)* credible mechanisms governing estimates of vibratory ground motion that can occur at a site, *(b)* a review of available historical and instrumental seismicity data (including strong motion data) to assess and calibrate the model, ~~and~~ *(c)* applicable (existing and/or newly developed) ground motion prediction equations for the ground motion estimates**, and** ***(d)* reference soil or rock horizon (defined by shear wave velocity, density, and damping values)**.
SHA-D2	--------------------	No objection	--------------------
SHA-D3	The ground motion characterization model should include ground motion prediction equations (GMPEs) with alternative distance and magnitude scaling behaviors, not just a range of amplitudes.	Clarification	ENSURE that uncertainties are included in the model that determine the range of seismic vibratory ground motion that can occur at a site **as well as alternative magnitude and distance scaling** ~~behaviors~~ in accordance with the level of analysis identified for HLR-SHA-A and the data and information in the update of the PSHA.
SHA-D4	--------------------	No objection	--------------------
Notes	--------------------	No objection	--------------------
Table 5-2.1-6			
Introductory text	--------------------	No objection	--------------------
SHA-E1, SHA-E2	--------------------	No objection	--------------------
SHA-E3	The term "ENSURE" is not the appropriate action verb.	Clarification	**JUSTIFY** ~~ENSURE that~~ the approach used to incorporate the site response analysis into the hazard analysis ~~is justified~~ **(e.g., sources of soils and rock material properties used in the analysis, uncertainties in site characterization and material properties, data to identify the depth to bedrock, appropriateness of one- two- or three-dimensional analysis in relation to the site stratigraphy)**.

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Index No	Issue	Position	Resolution
Table 5-2.1-7			
Introductory text	--------------------	No objection	--------------------
SHA-F1 through SHA-F3	--------------------	No objection	--------------------
Notes	--------------------	No objection	--------------------
Table 5-2.1-8			
Introductory text	--------------------	No objection	--------------------
SHA-G1, SHA-G2	--------------------	No objection	--------------------
Notes	--------------------	No objection	--------------------
Table 5-2.1-9			
Introductory text	--------------------	No objection	--------------------
SHA-H1, SHA-H2	--------------------	No objection	--------------------
Notes	--------------------	No objection	--------------------
Table 5-2.1-10			
Introductory text	--------------------	No objection	--------------------
SHA-I1	--------------------	No objection	--------------------
SHA-I2	The supporting requirement uses the terms "hazards" and "secondary hazard" interchangeably, which is potentially confusing.	Clarification	For those **secondary** hazards that are not screened out, INCLUDE their effect through assessment of the frequency of ~~hazard~~ occurrence and ~~the~~ magnitude**, when applicable,** of the secondary hazard.
Note 2	The last sentence of Note (2) in Table 5-2.1-10 is vague and unnecessary.	Clarification	The appropriate approach used to justify the basis and methodology used for screening out secondary hazards is hazard- and site-specific. ~~Justification may be based on available public literature and prior hazard studies.~~

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Index No	Issue	Position	Resolution
Table 5-2.1-11			
Introductory text	--------------------	No objection	--------------------
SHA-J1, SHA-J2	--------------------	No objection	--------------------
Notes	--------------------	No objection	--------------------
Section 5-2.2			
Introductory text	--------------------	No objection	--------------------
Table 5-2.2-1			
HLR-SFR-A through HLR-SFR-F	--------------------	No objection	--------------------
Table 5-2.2-2			
Introductory text	--------------------	No objection	--------------------
SFR-A1	The intent of supporting requirement SFR-A1 needs additional clarification.	Clarification	The NMA already discusses the overall intent of SFR-A1 and distinguishes between failure mechanism and failure mode. Include in the NMA a discussion such as the following: **The intent of SFR-A1 is to ensure that the fragility analyst provides fragility assessments for the SSCs defined by the systems analyst in the plant's SEL and for the relevant failure modes associated with the basic PRA events. The understanding is that fragility assessments relate to failure mechanisms, which, in turn, relate to failure modes defined by the systems analyst.**
SFR-A2	The information to be included should be such that it can justify the modeling of SSCs as correlated from a fragility perspective and not	Clarification	INCLUDE information ~~relevant to~~ **justifying the** modeling of fragility **dependency** ~~correlation~~ of SSCs and its basis (e.g., similarity of component construction and location, **and response**

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	simply be relevant. Justification, more than the examples provided, will be necessary for any correlation other than 0 and 1. Additionally, the phrase “fragility correlation” should be replaced with “fragility dependence.” Dependence between random variables characterizes their interrelationship. Correlation (coefficient) is used to define the dependence structure between random variables. It is also lacking criteria for determining the acceptability of a correlation model.		**spectra at the locations**) to support SPR-B4.
Table 5-2.2-3			
Introductory text	--------------------	No objection	--------------------
SFR-B1, through SFR-B3	--------------------	No objection	--------------------
SFR-B4	The action verb ESTIMATE implies using judgement or qualitative measures, which are inconsistent with the intent of the SR. The action verb CALCULATE involves a mathematical process, whereas the action verb ESTIMATE does not necessarily involve a calculation (e.g., quantification of a probability or frequency)	Qualification	If median-centered response analysis is performed, **CALCULATE** ~~ESTIMATE~~ the median response (i.e., structural loads and floor response spectra) and **ESTIMATE the** variability in the response.

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Index No	Issue	Position	Resolution
	and can be derived qualitatively.		
SFR-B5	--------------------	No objection	--------------------
SFR-B6	In the 2009 revision (SFR-C2), part of the SR asked one to ACCOUNT for the entire spectrum of input ground motion levels displayed in the seismic hazard curves. This sentence is removed in the Code Case. However, this sentence also ensures the quality of the results of the probabilistic response analysis.	Qualification	If probabilistic response analysis is performed to calculate structural loads and floor response spectra, ENSURE that the number of simulations done (e.g., Monte Carlo simulation or Latin Hypercube Sampling) is large enough to calculate stable responses. **ACCOUNT for the entire spectrum of input ground motion levels displayed in the seismic hazard curves.**
Table 5-2.2-4			
Introductory text	--------------------	No objection	--------------------
SFR-C1	The intent is to provide the basis and methodology to justify that the capacity of the SSC exceeds the screening level.	Clarification	SPECIFY the basis and methodologies established for the capacity-based screening for the level defined in SPR-B5 **(e.g., use of simplified fragility analysis, use of applicable generic fragility or qualification data or earthquake experience, and use and applicability of EPRI fragility screening guidance are examples)**.
SFR-C2	In ASME/ANS RA-Sa-2009, Note (2) of the corresponding supporting requirement (i.e., SFR-B2) indicates that the screening criteria do not apply to high-seismic regions such as coastal California. However, SFR-C2 in the Code Case does not discuss this note.	Qualification	~~SPECIFY~~ **JUSTIFY** the basis for screening of inherently rugged components **(e.g., applicability of fragility or qualification test data, earthquake experience, past fragility analysis for similar SSCs and seismic responses, applicable EPRI guidance)**.
Table 5-2.2-5			

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Index No	Issue	Position	Resolution
Introductory text	--------------------	No objection	--------------------
SFR-D1, SFR-D2	--------------------	No objection	--------------------
SFR-D3	For Capability Category (CC) I: In general, the walkdown AND the fragility evaluation provide the assurance. This requirement supports that assurance but may not always ensure. Also, "vulnerability" needs to be defined.	Clarification	IDENTIFY ~~seismic vulnerabilities~~ **low seismic capacities** ~~and~~ **to ensure** ~~ENSURE~~ that assumptions and the use of generic seismic fragilities are conservative.
SFR-D3	For CC II: The current language implies realistic and plant-specific fragilities for all vulnerabilities, which is inconsistent with SFR-E3 and established practice.	Clarification	IDENTIFY seismic vulnerabilities to ensure ~~and ENSURE~~ that the seismic fragility calculations can be realistic and plant-specific as needed.
SFR-D4	The walkdown should also focus on operator pathways and potential unavailability of those pathways. SFR-D7 seems to refer to consequences of failure of one SSC on the performance of another SSC, including inoperability of the SSC by an operator action. However, the words added here refer to pathways for ex-control room actions.	Clarification	FOCUS on potential functional and structural failure modes, equipment anchorage, ~~and~~ support load paths, **and pathways necessary for performing required ex-control room actions**.
SFR-D5	--------------------	No objection	--------------------
SFR-D6	In ASME/ANS RA-Sa-2009, SFR-E3 indicates that if a component is screened out during or following	Qualification	Add the following or equivalent as a new SFR-D6: IDENTIFY credible seismic-induced failure for the fire sources provided in

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Index No	Issue	Position	Resolution
	the walkdowns, document the anchorage calculation and provide the basis. However, this statement is removed in the Code Case, and it is not clear whether screening out equipment during walkdowns is allowed.		SPR-C4. **If components are screened out during or following the walkdown, PROVIDE the basis justifying such a screening.**
SFR-D7	This supporting requirement appears to prejudge which seismic interactions have the potential to be "risk-significant" prior to the walkdown. If the intent is that such information will be provided to the walkdown team by the plant-systems analyst, it appears to be premature to expect such information to be available at the time of walkdown. Further, such an intent or appearance of intent can lead to an argument for excluding the plant-systems analyst from the walkdown. The second part of the SR starting with "EVALUATE the consequences…" is expected to capture the "risk-importance" of the identified interactions.	Clarification	IDENTIFY ~~potential risk-significant~~ **credible** seismic interactions including proximity impacts, falling hazards, and differential displacements (e.g., failure and falling of masonry walls and nonseismically designed SSCs, impact between cabinets, differential building displacements), and EVALUATE the consequences of such interactions on SSCs contained in the systems model and on the credited operator actions. (See HLR-SPR-D.)
Table 5-2.2-6			
Introductory text	--------------------	No objection	--------------------
SFR-E1	--------------------	No objection	--------------------
SFR-E2	For CC I: The intent of the requirements should be to identify relevant failure mechanisms. In	Qualification	For SSCs identified in ~~SPR-C4~~ **SPR-C6** that significantly contribute to seismic core damage frequency and/or seismic large early release frequency,

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Index No	Issue	Position	Resolution
	CCI conservative assumptions and data may be used.		~~conservatively~~ IDENTIFY relevant failure ~~modes~~ **mechanisms** of structures, equipment, and soil. **ENSURE that the assumptions and data used in the identification are conservative.**
SFR-E2	For CC II: The examples listed in the requirement confuse the differences between CC I and CC II. The only real difference is that CC I states "conservatively IDENTIFY relevant" while CC II says "IDENTIFY relevant and realistic". This SR references SPR-C4 but should reference SPR-C6.	Clarification	For those SSCs identified in ~~SPR-C4~~ **SPR-C6** that significantly contribute to seismic core damage frequency and/or seismic large early release frequency, IDENTIFY relevant and realistic failure **mechanisms** ~~modes~~ of structures, equipment, and soil.
SFR-E3	For CC I: Seismic fragility estimates should be conservative. Ensuring the fragility estimate is conservative may require the development of a more realistic estimate to compare against, which arguably makes the CC I requirement more of a CC II requirement and effectively defeats the purpose of establishing a CC I requirement. Therefore, besides the action verb, the main distinction between CC I and CC II should be the "refinement" of fragilities (conservative or bounding for CC I and realistic for CC II).	Qualification	ESTIMATE **conservative** seismic fragilities for the failure modes of interest identified in SFR-E2 using plant-specific data, ~~and ENSURE that they are realistic.~~ **or** JUSTIFY (e.g., through the calculation of seismic CDF and LERF per HLR-SPR-E) the use of generic fragility data (e.g., fragility test data, generic seismic qualification test data, and earthquake experience data) or ~~conservative~~ assumptions for the SSCs as being **conservative** ~~appropriate for the plant and not significant to the overall results~~.
SFR-E3	For CC II: Calculated seismic fragility should be realistic. The requirement to ensure is unnecessary as this effectively is a function of a peer review.	Qualification	CALCULATE **realistic** seismic fragilities for the failure modes of interest identified in SFR-E2 using plant-specific data, ~~and ENSURE that they are realistic.~~ **or** JUSTIFY (e.g., through the calculation of seismic CDF and LERF per HLR-SPR-E)

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Index No	Issue	Position	Resolution
	The addition of showing no difference in insights or masking of risk helps ensure the use of generic fragility or conservative assumptions is acceptable for CC II.		the use of generic fragility data (e.g., fragility test data, generic seismic qualification test data, and earthquake experience data) or conservative assumptions for any SSCs **by showing no differences in insights or masking of risk** ~~as being appropriate for the plant~~.
SFR-E4	--------------------	No objection	--------------------
SFR-E5	The SR (CC I and II) refers to SPR-B6 for identification purposes. SPR-B6 discusses “relay or other similar devices.” To capture these items, this SR also needs to apply to “other similar devices” to prevent any implication that “other similar devices” need not be considered here. Additionally, the action verb for the second part of the CC II requirements needs to be capitalized to identify it.	Clarification	For CCI: ESTIMATE contact-chatter seismic fragilities for relays **or other similar devices** that are identified in the systems analysis. (See SPR-B6.) For CCII: CALCULATE contact-chatter seismic fragilities for relays **or other similar devices** that are identified in the systems analysis (see SPR-B6) that significantly contribute to seismic core damage frequency and/or seismic large early release frequency.
SFR-E6	For CC II: The action verb for the second part of the CC II requirements needs to be capitalized to identify it. “Calculate” is the appropriate action verb to be used for this supporting requirement.	Qualification	CALCULATE seismic fragilities for credible seismic-induced flood sources (see SFR-D5) and seismic-induced fire sources (see SFR-D6) that significantly contribute to seismic core damage frequency and/or seismic large early release frequency. For those flood and fire sources that do not significantly contribute to seismic core damage frequency and/or seismic large early release frequency, ~~estimate~~ the seismic fragilities.
Table 5-2.2-7			
Introductory text	--------------------	No objection	--------------------
SFR-F1	--------------------	No objection	--------------------
SFR-F2	Related Table 5-2.2-6 that provides supporting requirements associated	Clarification	Regarding list item *(i)* in SFR-F2:

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	with the calculation of seismic-fragility parameters use distinct action verbs ESTIMATE and CALCULATE, respectively, for CC I and CC II. However, the related supporting requirement SFR-F2, item *(i),* associated with documentation of fragility parameter values only uses the word "estimation" but not "calculation." Therefore, the documentation supporting requirement item *(i)* is partly inconsistent with other related supporting requirements.		*(i)* estimation **or calculation** of fragility parameter values for each SSC modeled (median capacity, logarithmic standard deviation reflecting the randomness in median capacity, and logarithmic standard deviation representing the uncertainty in median capacity), and
SFR-F3	--------------------	No objection	--------------------
Section 5-2.3			
Introductory text, Issue 1	The seismic PRA depends on both the capability and completeness of the internal events at-power PRA.	Clarification	It is assumed: • Relative to the systems-analysis requirements contained herein, the seismic PRA analysis team possesses a full-scope internal events, at-power Level 1 and Level 2 LERF PRA, developed either before or concurrently with the seismic PRA. • The internal-events PRA is then used as the basis for the seismic PRA systems analysis. It is recognized that the capability and completeness of the seismic PRA is a function of the capability **and completeness** of the internal events at-power PRA.
Introductory text, Issue 2	The sentence reads like a "how to," which is not the intent of the standard. Further, none of the references cited in the section are endorsed by	Clarification	~~A general methodology for the modeling and quantification of a seismic PRA is documented in references such as EPRI-3002000709 [5-5], EPRI-1020756 [5-6], and EPRI-1025294 [5-7].~~

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	the staff. Such references should be moved to the NMA portion of the standard.		
Introductory text, Issue 3	It needs to be ensured that cross-references in SFR SRs to SPR SRs are also cross-referenced in the related SPR SRs. For example: - SPR-B4 includes the reference to SFR-A2. - SPR-B5 includes the reference to SFR-C1. - SPR-C4 does not cross-reference SFR-D6. - SPR-D does not cross-reference SFR-D7.	Clarification	Include the missing cross-references either in the requirements or the footnotes.
Table 5-2.3-1			
HLR-SPR-A through HLR-SPR-C	--------------------	No objection	--------------------
HLR-SPR-D	The term "operator performance" can be interpreted in a narrow context to mean only in-control room actions and performance. However, the HLR and the corresponding SRs are applicable to all human actions included in the seismic PRA.	Clarification	Human actions credited in the seismic PRA shall consider seismic-specific challenges to ~~operator performance~~ **actions included in the seismic PRA.**
HLR-SPR-E	--------------------	No objection	--------------------
HLR-SPR-F	This HLR is overly broad, since HLR-SHA-J and HLR-SFR-F already address documentation of the seismic hazard evaluation and the seismic fragility evaluation, respectively.	Clarification	Documentation of the seismic ~~PRA analysis~~ **plant-response model** shall be consistent with the applicable supporting requirements.

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Index No	Issue	Position	Resolution
Table 5-2.3-2			
Introductory text	--------------------	No objection	--------------------
SPR-A1	--------------------	No objection	--------------------
SPR-A2	It is unclear whether the SR is seeking to identify all possible initiating events from secondary hazards or if the intent is to identify and screen such initiators for inclusion in the plant-systems model.	Clarification	Using a systematic process, IDENTIFY **credible** seismically induced initiating events caused by secondary hazards (e.g., seismically induced internal flooding, external flooding, and fire) including those identified in SHA-I2 **for consideration** ~~retention~~ **in the plant-response analysis and model development process**.
SPR-A3	The verb "encompasses" is overly severe and cannot reasonably be achieved in practice. The wording of this SR should be similar to that of IE-A3 and IE-A4.	Clarification	REVIEW plant-specific response to past seismic events, as well as other available seismic risk evaluations for nuclear plants, to ensure that the list of initiating events included in the evaluation ~~encompasses~~ **accounts for** industry experience.
SPR-A4	The plant-response analysis should include all identified events.	Clarification	INCLUDE in the plant-response analysis the events identified in SPR-A1**,** ~~and~~ SPR-A2**, and SPR-A3** above.
Notes	--------------------	No objection	--------------------
Table 5-2.3-3			
Introductory text	--------------------	No objection	--------------------
SPR-B1	--------------------	No objection	--------------------
SPR-B2	Due to the input from the fire and internal flooding PRAs, and possibly other hazard PRAs, in addition to internal events the findings from all relevant PRAs should be appropriately dispositioned. Additionally, it is not clear what is intended by the latter part of this SR	Qualification	ENSURE that the peer review findings for the internal-events **and other hazard** PRA**s** that are relevant to the seismic PRA are resolved and ~~that the disposition does not adversely affect~~ **into** the development of the seismic PRA plant-response model.

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Index No	Issue	Position	Resolution
	("…does not adversely affect…").		
SPR-B3	Incorrect reference to SPR-C4 instead of SPR-C6.	Clarification	INCLUDE seismically induced failures representing the failure modes of interest in the seismic PRA plant-response model (e.g., tank rupture, pump failure to start/run, etc.). (See SPR-C4 **SPR-C6**.)
SPR-B4	--------------------	No objection	--------------------
SPR-B5	The justification for the appropriate capacity-based screening level needs to be provided. Neither the action verb for this SR nor that used for SFR-C1 achieves that purpose.	Qualification	~~SPECIFY~~ **JUSTIFY (e.g. based on the contribution to the risk quantification)** ~~an appropriate~~ **the** set of criteria to be used in support of the screening of SSC failure modes on the basis of fragility. (See SFR-C1.)
SPR-B6	The term "with a significant contributor to CDF or LERF" is not defined. How can one determine the significance without performing the calculation?	Clarification	USE a systematic approach to INCLUDE in the system analysis the effects of those relays or similar devices **susceptible to contact chatter** whose contact chatter results in the unavailability or spurious actuation of SSCs **on the seismic equipment list.** ~~with a significant contribution to CDF or LERF.~~
SPR-B7 through SPR-B11	--------------------	No objection	--------------------
Notes	--------------------	No objection	--------------------
Table 5-2.3-4			
Introductory text	--------------------	No objection	--------------------
SPR-C1 through SPR-C6	--------------------	No objection	--------------------
Table 5-2.3-5			
Introductory text	--------------------	No objection	--------------------
SPR-D1, SPR-D2	--------------------	No objection	--------------------

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SPR-D3	Cue availability as well as dependencies are integral part of human reliability analyses and maybe affected by seismic events.	Clarification	For CC I: CALCULATE the HEPs for all HFEs taking into account relevant seismic-related effects on control room and ex-control room post-initiator actions in accordance with the SRs for HLR-HR-G in Part 2 of this Standard as set forth under Capability Category I. In addressing influencing factors and the timing considerations covered in SRs HR-G3, HR-G4, and HR-G5 in Part 2, attention is to be given to how the seismic event alters any previous assessments in non-seismic analyses including: additional workload and stress; effects of the seismic event on mitigation, **cue availability**, **dependencies,** required response, timing, accessibility, and potential for physical harm; and seismic-specific job aids and training. For CC II: CALCULATE the HEPs for all HFEs taking into account relevant seismic-related effects on control room and ex-control room post-initiator actions in accordance with the SRs for HLR-HR-G in Part 2 of this Standard as set forth under Capability Category II. In addressing influencing factors and the timing considerations covered in SRs HR-G3, HR-G4, and HR-G5 in Part 2, attention is to be given to how the seismic event alters any previous assessments in non-seismic analyses including: additional workload and stress; effects of the seismic event on mitigation, **cue availability**, **dependencies,** required response, timing, accessibility, and potential for physical harm; and seismic-specific job aids and training.
SPR-D4	The action verb ESTIMATE implies using judgement or qualitative measures only, which is inconsistent with the intent of the SR. Some of	Qualification	For significant HFEs, ~~ESTIMATE~~ **DETERMINE** the timing aspects of the response actions (i.e., time of relevant indication, time available to complete action, and time required to complete action) recognizing the sequence of events

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Index No	Issue	Position	Resolution
	the examples of approaches provide more information than an estimate.		and expected seismic conditions based on one or a combination of the following approaches: *(a)* Walk-throughs or talk-throughs of procedures with plant operations or training personnel *(b)* Simulator observations *(c)* Plant-specific thermal-hydraulic analyses *(d)* Realistic and applicable generic or similar plant thermal-hydraulic analyses. Based on a review of procedures with plant operations or training personnel and recognizing the sequence of events and expected seismic conditions, CONFIRM for nonsignificant HFEs the timing aspects of the response actions.
SPR-D5	--------------------	No objection	--------------------
Notes	--------------------	No objection	--------------------
Table 5-2.3-6			
Introductory text	--------------------	No objection	--------------------
SPR-E1 through SPR-E3	--------------------	No objection	--------------------
SPR-E4	The phrase “dominant sequence insights” is not defined in either Addendum A or Addendum B. The term “dominant” was intentionally not used anywhere in the standard.	Clarification	USE the quantification process to ensure that the components screened out, based on the screening level defined in SPR-B5, do not become a significant contributor or do not invalidate the ~~dominant~~ **significant** sequence insights of the seismic PRA.
SPR-E5	For CC II: It is not possible or necessary to quantify all uncertainties.	Clarification	For CC II: QUANTIFY the mean core damage frequency and large early release frequency and propagate the **parameter** uncertainty that results from each input (i.e., the seismic hazard, the seismic fragilities, and the systems analysis).

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Index No	Issue	Position	Resolution
SPR-E6	--------------------	No objection	--------------------
SPR-E7	For CC II: The reference to Part 2 is missing for HLR-QU-E for CC II.	Clarification	For CC II: PERFORM the uncertainty analysis consistent with HLR-QU-E **of Part 2** addressing key assumptions in the hazard analysis (see SHA-J2), fragility analysis (see SFR-F3), and system modeling for Capability Category II.
Notes	--------------------	No objection	--------------------
Table 5-2.3-7			
Introductory text	--------------------	No objection	--------------------
SPR-F1	--------------------	No objection	--------------------
SPR-F2	The Code Case needs to specify the type of documentation to be provided, rather than relying on the discretion of the user.	Clarification	DOCUMENT the process used in the seismic plant-response analysis and quantification**, including**~~. For example, this documentation typically includes a description of~~
SPR-F3	--------------------	No objection	--------------------
Notes	--------------------	No objection	--------------------
Section 5-3			
Section 5-3	--------------------	No objection	--------------------
Section 5-4			
Section 5-4	--------------------	No objection	--------------------
Nonmandatory Appendix 5-A			
Nonmandatory Appendix 5-A	--------------------	No objection	--------------------

REFERENCES [13]

1. ASME/ANS Standard ASME/ANS RA-S Case 1, "Case for ASME/ANS RA-Sb-2013 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment of Nuclear Power Plant Applications," ASME, New York, NY, November 22, 2017. [14]

2. American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RA-S-2008, ASME, New York, NY, ANS, La Grange Park, Illinois, February 2009.

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13 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

14 Copies of ASME standards may be purchased from ASME, Two Park Avenue, New York, New York 10016-5990; telephone (800) 843-2763. Purchase information is available through the ASME Web-based store at http://www.asme.org/Codes/Publications/.

APPENDIX C

GUIDANCE FOR CLASSIFYING CHANGES TO A PROBABILISTIC RISK ASSESSMENT AS PRA MAINTENANCE OR A PRA UPGRADE

A change to a base probabilistic risk assessment (PRA) may involve PRA maintenance or a PRA upgrade, including the use of a newly developed method (NDM). The distinction between these activities is important because a PRA upgrade, including the use of an NDM, should be peer reviewed, whereas changes due to PRA maintenance do not need to be peer reviewed. The level of detail of the peer review and the peer review team member qualifications differ between the peer review for a PRA upgrade and the peer review for the use of an NDM.

C-1 PRA Maintenance and PRA Upgrades

PRA maintenance activities do not need a peer review because the base PRA will have been previously peer reviewed and the licensee or applicant has not made fundamental changes to the base PRA. Consequently, the licensee or applicant will have demonstrated experience in applying the methods in the base PRA model. This is not true for PRA upgrades or use of an NDM, which have not been previously peer reviewed. Therefore, the licensee or applicant has not demonstrated experience in applying the changes to the base PRA.

In applying the guidance described in the staff position on peer reviews for PRA upgrades or NDMs (regulatory position C.2.2 of Section C of this regulatory guide (RG)), the licensee or applicant should determine whether the change to the PRA is PRA maintenance or a PRA upgrade. Since PRA maintenance is defined as a change to the PRA that does not meet the definition of a PRA upgrade, the classification process is focused on determining whether a change to the PRA is a PRA upgrade, which includes use of an NDM. The definitions of the terms "PRA upgrade," "state-of-practice," "PRA method," "consensus method," and "NDM" are provided in the Glossary of this RG. Consistent with these definitions, a change to the PRA is considered a PRA upgrade when the change results in the following:

1. application of a PRA method previously used in the peer reviewed plant PRA that either—
    a. causes one or more supporting requirements in a national consensus PRA standard to be applicable that were not applicable in the previous peer review (i.e., including changes in capability category), or
    b. is being applied in a different context.
2. application of a method not previously used in the peer reviewed plant PRA that either—
    a. is a consensus method or an NDM that has been subjected to an NDM peer review, or
    b. has been developed separately from a state-of-practice method or involves a fundamental change to a state-of-practice method (i.e., an NDM).

Changes related to list item 2.b require an NDM peer review, consistent with regulatory position C.2.2.2.2. Consistent with the above criteria for classifying a change to the PRA as PRA maintenance or a PRA upgrade and the definitions in the Glossary of this RG, the NRC staff endorses the guidance provided in Section 3 of PWROG-19027-NP, Revision 2, "Newly Developed Method Requirements and Peer Review," issued July 2020 (Ref. 1), as one acceptable approach for determining whether a change to a PRA model is classified as PRA maintenance or a PRA upgrade.

C-2 Newly Developed Method

An NDM is a method that has either been developed separately from a state-of-practice method or is one that involves a fundamental change to a state-of-practice method, with respect to use in a PRA. An NDM for a PRA may well be used by other industries or other applications, but has not been used to meet the requirements for a PRA model as defined by a national consensus PRA standard, as endorsed in this RG with exceptions and clarifications. Therefore, an NDM is not considered to be part of the state-of-practice in developing PRA models for a nuclear power plant. The NRC staff endorses the definition of the term "NDM" provided in PWROG-19027-NP, Revision 2, which has been reproduced in the Glossary of this RG for reference.

As indicated in the definition of an NDM, such a method may have been developed separately from an existing method. Although the purpose and goal of the NDM may be similar to that of an existing method, the NDM's technical bases (e.g., assumptions and data) and the tools (e.g., analyses and equations) used to formulate the method are fundamentally different than those of the existing method. Additionally, an NDM may have been developed from an existing method that has been modified. A modified existing method is considered an NDM when the modifications result in fundamental changes to the technical bases and tools used to formulate the method. In contrast, when a modified existing method does not result in such fundamental changes, it would not be considered an "NDM."

REFERENCES [15]

1. Pressurized Water Reactor Owners Group (PWROG), Report PWROG-19027-NP, Revision 2, "Newly Developed Method Requirements and Peer Review," Cranberry Township, PA, July 2020. (ADAMS Accession No. ML20213C660)

15 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.

APPENDIX D

HAZARDS FOR CONSIDERATION IN A BASE PROBABILISTIC RISK ASSESSMENT

A key feature of a base probabilistic risk assessment (PRA) is that a wide spectrum of potential hazards in terms of magnitude and frequency of occurrence should be systematically surveyed to help ensure that significant contributors to plant risk are not inadvertently excluded from the PRA. A hazard is a category of similar challenges to plant design or operations that poses some risk to a facility. A hazard group is a set of similar hazards that are assessed in a PRA using common approaches, methods, and likelihood data for characterizing the effect on the plant. Hazards represent events or phenomena that are generally classified as either internal hazards or external hazards, based on the defined plant boundary in a PRA. Hazards categorized under the internal events, internal flood, internal fire, seismic, high-wind, and external flood hazard groups are typically analyzed and modeled in detail using a PRA. However, there are a number of internal and external hazards whose risk to a facility can be assessed qualitatively, quantitatively, or both, but in a simplified manner and without the need for a detailed PRA model. Regulatory position C.1.2.6 in Section C of this regulatory guide (RG) provides additional guidance on screening and conservative analyses that can be performed to this end. Conversely, some such internal and external hazards may produce impacts to a plant and a potential plant response that are too complex to be represented by a simplified analysis and should be modeled in detail using a PRA. This latter type of hazard is commonly referred to as an “other hazard” and regulatory position C.1.2.9 provides additional guidance on the modeling of such hazards.

A list of hazards and their potential impacts that should be considered include, but may not be limited to, those items listed in Tables D-1 and D-2. Table D-1 provides a list with general descriptions of the hazard groups and the hazards within those groups that should be considered during the development of a base PRA, consistent with the list of hazards provided in Appendix 6-A of American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications” (Ref. 1). The genesis of this list can be traced back to NUREG/CR-2300, “PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants,” issued January 1993 (Ref. 2), and earlier nuclear power plant PRA studies. This list of hazards has evolved and expanded over the past several decades based on insights and lessons learned from other PRA-related programs and applications such as licensees’ responses to Generic Letter 88-20, Supplement 4, “Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities—10 CFR 50.54(f),” dated June 28, 1991 (Ref. 3). Table D-2 provides a list of hazard causes and potential conditions to consider during the process of determining whether a given hazard poses some risk to a facility. The taxonomy of these hazard groups and the hazards within those groups are relevant to PRA applications only.

Table D-1. List of Hazards

Hazard Group/ Hazard	Direct or Secondary Impact of Hazard
Aircraft Impact	An aircraft (either a portion of (e.g., missile) or the entire aircraft) that collides either directly or indirectly (i.e., skidding impact with one or more structures, systems, or components (SSCs) at or in the plant’s analyzed area causing functional failure. Secondary hazards resulting from an aircraft impact include, but are not necessarily limited to, fire.

Hazard Group/ Hazard	Direct or Secondary Impact of Hazard
Avalanche	Rapid flow of a large mass of accumulated frozen precipitation and other debris down a sloped surface resulting in dynamic loading of SSCs at or in the plant's analyzed area causing functional failure or adverse impact on natural water supplies used for heat rejection.
Biological Events	Accumulation or deposition of vegetation or organisms (e.g., zebra mussels, clams, fish, algae) on an intake structure or internal to a system that uses raw cooling water from a source of surface water, causing its functional failure.
Coastal Erosion	Removal of material from a shoreline of a body of water (e.g., river, lake, ocean) due to surface processes (e.g., wave action, tidal currents, wave currents, drainage, or winds and including river bed scouring) that results in damage to the foundation of SSCs at or in the plant's analyzed area, causing functional failure.
Drought	A shortage of surface water supplies due to a period of below-average precipitation in a given region, thereby depleting the water supply needed for the various water-cooling functions at the facility.
External Flood	An excess of water outside the plant boundary that causes functional failure to plant SSCs. External flood causes include, but may not be limited to, flooding due to dam failure, high tide, hurricane (tropical cyclone), ice cover, local intense precipitation, river diversion, river and stream overflow, seiche, storm surge, and tsunami.
Extreme Winds and Tornadoes	Strong winds resulting in dynamic loading or missile impacts on SCCs causing functional failure. Hazards that could potentially result in high wind include the following: • hurricane—severe winds developed from a tropical depression resulting in missiles or dynamic loading on SSCs. Secondary hazards resulting from a hurricane, include, but are not necessarily limited to tornado • straight wind—a strong wind resulting in missiles or dynamic loading on SSCs that is not associated with either hurricanes or tornadoes • tornado—a strong whirlwind that results in missiles or dynamic loading on SSCs
Fog	Low-lying water vapor in the form of a cloud or obscuring haze of atmospheric dust or smoke resulting in impeded visibility that could result in, for example, a transportation accident.
Forest Fire	Direct (e.g., thermal effects) and indirect effects (e.g., generation of combustion products, transport of firebrand) of a forest fire outside the plant boundary that causes functional failure of plant SSCs. Hazards that could cause or be caused by a forest fire include, but may not be limited to, wildfires and grass fires.
Frost	A thin layer of ice crystals that form on the ground or the surface of an earthbound object when the temperature of the ground or surface of the object falls below freezing.
Hail	A shower of ice or hard snow that could result in transportation accidents or directly causes dynamic loading or freezing conditions as a result of ice coverage.

Hazard Group/ Hazard	Direct or Secondary Impact of Hazard
High Summer Temperature	Effects on SSC operation due to abnormally high ambient temperatures resulting from weather phenomena. Secondary hazards resulting from high ambient temperatures, include, but are not necessarily limited, to low lake or river water levels.
High Tide	The periodic maximum rise of sea level resulting from the combined effects of the tidal gravitational forces exerted by the Moon and Sun and the rotation of the Earth.
Hurricane (Tropical Cyclone)	Flooding that results from the intense rain fall from a hurricane (tropical cyclone). Secondary hazards resulting from a hurricane include, but are not necessarily limited to, dam failure, high tide, river and stream overflow, seiche, storm surge, and waves.
Ice Cover	Flooding due to downstream blockages of ice on a river. Secondary hazards resulting from an ice blockage include, but are not necessarily limited to, river and stream overflow.
Industrial or Military Facility Accident	An accident at an offsite industrial or military facility that results in a release of toxic gases, a release of combustion products, a release of radioactivity, an explosion, or the generation of missiles.
Internal Flood	Flooding that results from leaks or ruptures of liquid systems (e.g., tanks, pipes, valves, pumps) originating inside the defined plant site boundary.
Landslide	Dynamic loading of SSCs or impacts on natural water supplies used for heat rejection due to the movement of rock, soil, and mud down a sloped surface (does not include frozen precipitation).
Lightning	Effects on SSCs due to a sudden electrical discharge from a cloud to the ground or Earth-bound object.
Low Lake or River Water Level	A decrease in the water level of the lake or river used for power generation.
Low Winter Temperature	Effects on SSC operation due to abnormally low ambient temperatures resulting from weather phenomena. Secondary hazards resulting from low ambient temperatures include, but are not necessarily limited to, frost, ice cover, and snow.
Meteorite/Satellite Strikes	A release of energy due to the impact of a space object such as a meteoroid, comet, or human-caused satellite falling within the Earth's atmosphere, a direct impact with the Earth's surface, or a combination of these effects. This hazard is analyzed with respect to direct impacts of an SSC and indirect impact effects such as thermal effects (e.g., radiative heat transfer), overpressure effects, seismic effects, and the effects of ejecta resulting from a ground strike.
Pipeline Accident	A release of hazardous material, a release of combustion products, an explosion, or the generation of missiles due to an accident involving the rupture of a pipeline carrying hazardous materials.
Precipitation, Intense	Flooding that results from local intense precipitation. Secondary hazards resulting from local intense precipitation, include, but are not necessarily limited to, dam failure and river and stream overflow.

Hazard Group/ Hazard	Direct or Secondary Impact of Hazard
Release of Chemicals from Onsite Storage	A release of hazardous material including, but not limited to liquids, combustion products, or radioactivity. Such releases may be concurrent with or induce an explosion or the generation of missiles. In this context, an onsite release of radioactivity is assumed to be associated with low-level radioactive waste.
River Diversion	The redirection of all or a portion of river flow by natural causes (e.g., a riverine embankment landslide) or intentionally (e.g., power production, irrigation).
Sandstorm	Persistent heavy winds transporting sand or dust that infiltrate SSCs at or in the plant's analyzed area causing functional failure.
Seiche	Flooding from water displaced by an oscillation of the surface of a landlocked body of water, such as a lake, that can vary in period from minutes to several hours.
Seismic Activity [See Note 1]	Sudden ground motion or vibration of the Earth as produced by a rapid release of stored-up energy along an active fault. Secondary hazards resulting from seismic activity include, but are not necessarily limited to, avalanche (both rock and snow), dam failure, industrial accidents, landslide, seiche, tsunami, and vehicle accidents.
Snow	The accumulation of snow could result in transportation accidents or directly cause dynamic loading or freezing conditions as a result of snow cover.
Soil Shrink-Swell	Dynamic forces on structures' foundations due to the expansion (swelling) and contraction (shrinking) of soil resulting from changes in the soil moisture content.
Storm Surge	Flooding that results from an abnormal rise in sea level due to atmospheric pressure changes and strong wind generally accompanied by an intense storm. Secondary hazards resulting from a storm surge include, but are not necessarily limited to, high tide, river and stream overflow, and waves.
Toxic Gas	A release of hazardous toxic or asphyxiant gases. Such releases may be concurrent with or induce an explosion or the generation of missiles. In this context, an onsite release of radioactivity is assumed to be associated with low-level radioactive waste.
Transportation Accidents	Accidents involving transportation resulting in collision with SSCs, a release of hazardous materials or combustion products, an explosion, or a generation of missiles causing functional failure of SSCs. Hazards that could potentially result in transportation accidents include, for example, a vehicle, railcar or ship (boat) accident that involves a collision or derailment, potentially resulting in fire, explosions, toxic releases, missiles, or other hazardous conditions.
Tsunami	Flooding that results from a series of long-period sea waves that displaces massive amounts of water as a result of an impulsive disturbance, such as a major submarine slides or landslide. Secondary hazards resulting from a tsunami include, but are not necessarily limited to, river and stream overflow.

Hazard Group/ Hazard	Direct or Secondary Impact of Hazard
Turbine-Generated Missiles	Damage to safety-related SSCs from a missile generated internal or external to the plant PRA boundary from rotating turbines or other external sources (e.g., high-pressure gas cylinders). Damage may result from a falling missile or a missile ejected directly toward safety-related SSCs (i.e., low-trajectory missiles).
Volcanic Activity [See Note 1]	Opening of Earth's crust resulting in tephra (i.e., rock fragments and particles ejected by volcanic eruption), lava flows, lahars (i.e., mud flows down volcano slopes), volcanic gases, pyroclastic flows (i.e., fast-moving flow of hot gas and volcanic matter moving down and away from a volcano), and landslides. Indirect impacts include distant ash fallout (e.g., tens to potentially thousands of miles away). Secondary hazards resulting from volcanic activity, include, but are not necessarily limited to, seismic activity and fire.
Waves	An area of moving water that is raised above the main surface of a body of water as a result of the wind blowing over an area of fluid surface.
[Note 1]	Seismic activity and volcanic activity are hazards in the tectonic activity hazard group. However, historically, seismic activity has been evaluated as a hazard group (Part 5 of this standard). Although seismic events due to volcanic activity may be evaluated as part of the seismic hazard group, other potential impacts of volcanic activity may need to be analyzed separately.

Table D-2. List of Hazard Causes and Conditions

Combustion/fire—resulting in burning, release of hot/toxic gases, release of combustion products, or heat causing functional failure of SSCs, the ability of the operator to perform, or both.
Debris effects—resulting in clogging of filters or adversely affecting equipment performance.
Dynamic forces (from dynamic or static loading)—resulting in structural damage to SSCs causing functional failure of SSCs or impeded operator ability to perform, or both
Explosions—resulting in fire, missiles, or gas releases
Effects on operator ability to do the following: • perform an action due to physical obstacles • perform a cognitive function • see • breathe • communicate • obtain available information (e.g., poor procedures, poor or no indication)
High energy arcs
Missiles—projectiles damaging structures and equipment
Physical obstruction—movement of structures or equipment reducing accessibility

Reduced air quality—combustion products affecting equipment or operator performance
Reduced availability of cooling water
Structural failure, including the following: • collapse • functional failure (e.g., break in containment, settlement of a structure) • loss of structural integrity
Thermal effects including the following: • heat transfer (radiative, conductive, or convective; advection (i.e., bulk transport of a fluid)) • steam
Water effects—water infiltration, submergence or spray causing corrosion, loss of electrical integrity (e.g., electrical short), clogging, inaccessibility, structural failure

REFERENCES [16]

1. American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RA-S-2008, ASME, New York, NY, ANS, La Grange Park, Illinois, February 2009.

2. U.S. Nuclear Regulatory Commission (NRC), "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," NUREG/CR-2300, Washington, DC, January 1983. (Agencywide Documents Access and Management System (ADAMS) Accession Package No. ML063560440)

3. NRC, Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities—10 CFR 50.54(f) (Generic Letter No. 88-20, Supplement 4)," Washington, DC, June 28, 1991. (ADAMS Accession No. ML031150485)

16 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public Web site at http://www.nrc.gov/reading-rm/doc-collections/ and through the NRC's Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html The documents can also be viewed online or printed for a fee in the NRC's Public Document Room (PDR) at 11555 Rockville Pike, Rockville, MD. For problems with ADAMS, contact the PDR staff at 301-415-4737 or (800) 397-4209; fax (301) 415-3548; or e-mail pdr.resource@nrc.gov.