U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.247

(For Trial Use)

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ACCEPTABILITY OF PROBABILISTIC RISK ASSESSMENT RESULTS FOR NON-LIGHT-WATER REACTOR RISK-INFORMED ACTIVITIES

A. INTRODUCTION

Purpose

This trial regulatory guide (RG) describes one acceptable approach that the U.S. Nuclear Regulatory Commission (NRC) staff has developed for determining whether a design-specific or plant-specific probabilistic risk assessment (PRA) used to support an application is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for non-light-water reactors (NLWRs). In this RG, the term "application" includes initial licensing applications and risk-informed applications. When used in support of an application, this RG will help reduce the need for an in-depth review of the PRA by the NRC and allow them to focus their review on key assumptions and areas identified as being of concern and relevant to the application and the demonstration of PRA acceptability.

The purpose of a trial RG is to allow adequate experience to be gained with the use and implementation of the guidance. Based on that experience and other potential changes to staff positions, the trial RG may be finalized after the trial use period to incorporate the related lessons learned, any comments received on the trial RG, or other pertinent information. This guide 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). Also, it endorses, with staff exceptions, a national consensus PRA standard provided by standards development organizations and industry guidance on PRA peer review.

Applicability

This RG applies to applications for NLWR licensing under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 2). Specific applicable regulations include the following:

- standard design certification (DC): 10 CFR Part 52, Subpart B
- combined license (COL): 10 CFR Part 52, Subpart C

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Electronic copies of this RG, 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.) ML21235A008. The regulatory analysis may be found in ADAMS under Accession No. ML21235A010.

- standard design approval (SDA): 10 CFR Part 52, Subpart E
- manufacturing license (ML): 10 CFR Part 52, Subpart F

This RG also applies to applications for NLWR licensing under 10 CFR Part 50, "Domestic licensing of production and utilization facilities" (Ref. 3). Specific applicable regulations include the following:

- construction permit (CP): 10 CFR Part 50
- operating license (OL): 10 CFR Part 50

Applicable Regulations

The following regulations are directly applicable to the use of PRA in licensing activities for NLWRs[1]:

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

    - 10 CFR 50.71(h)(2) requires that (1) each holder of a COL shall maintain and upgrade the PRA required by 10 CFR 50.71(h)(1), (2) the upgraded PRA must cover initiating events and modes of operation contained in NRC-endorsed consensus standards on PRA in effect 1 year before each required upgrade, and (3) 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 (ESPs), standard DCs, COLs, SDAs, and MLs 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 applicants for a standard DC under Subpart B of 10 CFR Part 52 shall describe the design-specific PRA and its results.

    - 10 CFR 52.79(a)(46) requires that applicants for a COL under Subpart C of 10 CFR Part 52 shall describe the plant-specific PRA and its results.

    - 10 CFR 52.79(c)(1) requires that applicants for a COL under Subpart C of 10 CFR Part 52 that reference an SDA under Subpart E of 10 CFR Part 52 shall use and update the PRA

[1] See the Background discussion in Section B of this RG for additional information about ongoing activities related to rulemakings for which the analysis described in this RG may inform applications to address those new or revised requirements.

information for the SDA to account for site-specific design information and any design changes or departures.

  - 10 CFR 52.79(d)(1) requires that applicants for a COL under Subpart C of 10 CFR Part 52 that reference a standard DC under Subpart B shall use and update the PRA information for the standard DC to account for site-specific design information and any design changes or departures.

  - 10 CFR 52.79(e)(1) requires that applicants for a COL under Subpart C of 10 CFR Part 52 that reference the use of one or more manufactured nuclear power reactors licensed under Subpart F of 10 CFR Part 52 shall use and update the PRA information for the manufactured reactor to account for site-specific design information and any design changes or departures.

  - 10 CFR 52.137(a)(25) requires that applicants for an SDA under Subpart E of 10 CFR Part 52 shall include a description of the design-specific PRA and its results.

  - 10 CFR 52.157(a)(31) requires that applicants for an ML under Subpart F of 10 CFR Part 52 shall include a description of the design-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 CP or OL applications (including requests for amendments) under 10 CFR Part 50 and ESP, DC, COL, SDA, and ML applications under 10 CFR Part 52 (including requests for amendments) for light water reactors (LWRs).

  - NUREG-0800, Section 19.1, “Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities” (Ref. 5), guides 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, Section 19.0, “Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors” (Ref. 6), pertains to the NRC staff review of the design-specific PRA for a DC and plant-specific PRA for a COL application.

- NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking,” issued March 2017 (Ref. 7), provides guidance on how to treat uncertainties associated with PRA in risk-informed decision-making. 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. 8), 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 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities” (Ref. 9), provides an acceptable approach for determining whether a base 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 decision-making for light-water reactors (LWRs).

- RG 1.206, "Applications for Nuclear Power Plants" (Ref. 10), provides guidance on the format and content of applications for nuclear power plants submitted to the NRC under 10 CFR Part 52, which specifies the information to be included in an application. This RG applies to power reactors with LWR technology. The NRC staff also considers this RG to generally apply to other types of power reactors (e.g., NLWRs). The NRC staff considers this guidance acceptable to support preparation of applications for ESPs, standard DCs, and COLs under 10 CFR Part 52 and generally acceptable to support its review of other types of applications under 10 CFR Part 52.

- RG 1.233, "Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology To Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light Water Reactors" (Ref. 11), provides guidance to inform the licensing basis and content of applications for NLWRs, including, but not limited to, molten salt reactors, high-temperature gas-cooled reactors, and a variety of fast reactors of different thermal capacities. NLWR applicants may use this guidance when applying for permits, licenses, certifications, and approvals under 10 CFR Part 50 and 10 CFR Part 52. RG 1.233 endorses the Licensing Modernization Project (LMP) guidance provided in Nuclear Energy Institute (NEI) 18-04, "Risk-Informed Performance-Based Technology Guidance for Non-Light Water Reactors" issued August 2019 (Ref. 12).

- 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. 13), provides guidance on how to adapt the American Society of Mechanical Engineers (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"[2] (Ref. 14), which was developed for currently operating reactors, to PRAs that support advanced LWR standard DC and COL applications.

Purpose of Regulatory Guides

The NRC issues RGs to describe methods that are acceptable to the staff for implementing specific parts of the agency's regulations, to explain techniques that the staff uses in evaluating specific issues 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 are acceptable if they provide a sufficient basis for the findings required for the issuance or continuance of a permit or license by the Commission. See section D for more information on the intended use of this draft regulatory guide.

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

[2] Because the PRA consensus standards use the terms "requirement," "require," and other similar mandatory language, the staff's endorsement, including staff exceptions, mirrors this language. However, the use of this language in this RG does not imply that this RG imposes any regulatory requirement or suggest that these standards are the only way to meet the statutory and regulatory requirements.

numbers 3150-0011 and 3150-0151, respectively. Send comments regarding this information collection to the FOIA, Library, and Information Collections 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, 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.

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B. DISCUSSION

Reason for Issuance

This RG provides guidance on one acceptable approach the NRC staff has developed for determining whether a design-specific or plant-specific PRA used to support an application is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for NLWRs. This RG is being issued consistent with OMB Circular A-119, "Federal Participation in the Development and Use of Voluntary Consensus Standards and in Conformity Assessment Activities" (Ref. 15), which directs Federal Government agencies and agency employees to use voluntary consensus standards, both domestic and international, in its regulatory and procurement activities in lieu of Government-unique standards, unless use of such standards would be inconsistent with applicable law or otherwise impractical. Accordingly, the NRC established processes for the review and endorsement of published voluntary consensus standards.

On February 8, 2021, ASME and ANS jointly published a voluntary consensus standard for NLWR PRA, ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants" (Ref. 16), referred to hereafter as the ASME/ANS NLWR PRA standard, which is endorsed in this RG with exceptions. In May 2021, NEI published industry PRA peer review guidance in NEI 20-09, Revision 1, "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard" (Ref. 17), which is endorsed in this RG with no exceptions.

Background

In 1995, the NRC issued "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement." This policy statement encourages the use of PRA in all regulatory matters and states, "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. 17), 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. 19). This document presented the staff's approach to defining the quality needed for pending or anticipated applications, as well as the process for achieving this quality, while allowing risk-informed decisions to be made using then available methods until all the necessary guidance documents were 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. 20), updated the staff's plan. Since issuance of the 1995 NRC policy statement, many applications 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, reliable, complete, and accurate and that it produces insights with appropriate fidelity to support anticipated risk-informed activities. As a result, the sufficiency of a PRA's technical content determines the acceptability of a PRA and its results. PRA acceptability describes the ability of a PRA to support risk-informed regulatory decision-making and is defined in terms of meeting the NRC regulatory positions in Section C of this RG, which can be satisfied by meeting the requirements of national consensus PRA standards and peer review processes, as endorsed by the NRC. Because consensus PRA standards use the terms "requirement," "require," and other similar mandatory language, the staff's endorsement, including exceptions, mirrors this language. However, the use of this language in this RG does not imply that the RG imposes any regulatory requirement or suggest that these standards are the only way to meet the statutory and regulatory requirements.

In general, national consensus PRA standards provide one set of minimum requirements that can be met, as endorsed by the NRC with exceptions, for a PRA to be considered acceptable. These consensus standards include both technical 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 PRA meets the requirements in the national consensus PRA standard. One acceptable approach for a peer review of a PRA is to have qualified personnel follow an established, NRC-endorsed peer review process that documents the results and identifies both strengths and weaknesses of the PRA. Use of this RG in support of an application will reduce the need for an in-depth review of the PRA by NRC reviewers, allowing them to focus on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application. The acceptability of a PRA is measured against the PRA scope, level of detail, conformance with the NRC regulatory positions in Section C of this RG, and representation of the modeled plant.

The ASME/ANS NLWR PRA standard provides requirements for a comprehensive probabilistic radiological risk assessment that addresses all radiological sources, all hazards, all plant operating states (POSs), and all levels of analysis (e.g., from initiating event to radiological consequence) of PRA for NLWRs. This RG provides the staff endorsement of the ASME/ANS NLWR PRA standard. Appendix A to this RG documents the bases for the staff's endorsement, with exceptions, of the technical requirements in the standard. The staff's endorsement, for trial use, with exceptions, is based on its review of the requirements in the ASME/ANS NLWR PRA standard against the related regulatory positions in Section C of this RG.

A PRA peer review process for NLWR PRAs was developed and documented in NEI 20-09, Revision 1, which provides guidance on how to perform a PRA peer review to meet the PRA peer review requirements in the ASME/ANS NLWR PRA standard. Regulatory Position C.2.2 of this RG provides guidance on the performance of PRA peer reviews and endorses NEI 20-09, Revision 1, in its entirety and with no exceptions as a means of satisfying the peer review requirements in the ASME/ANS NLWR PRA standard, as endorsed by the NRC in this RG. NEI 20-09, Revision 1, is based on a related industry PRA peer review guidance document, NEI 17-07, Revision 2, "Performance of PRA Peer Reviews Using the ASME/ANS PRA Standard," issued August 2019 (Ref. 21), as well as other earlier, related industry guidance documents. Reactor owners' groups and other industry organizations have been applying the PRA peer review process for several years, domestically and internationally. Consistent with the scope of the ASME/ANS NLWR PRA standard, NEI 20-09, Revision 1, addresses PRA peer reviews for an NLWR PRA that considers all radiological sources, all hazards, all POSs, and all levels of PRA analysis. Figure 1 illustrates the three, co-dependent aspects involved in achieving PRA acceptability.

Figure 1. NRC general framework for achieving PRA acceptability

Initial licensing application activities generally refer to any one or more of the types of applications listed in the Applicability section of this RG. The NRC staff notes that current regulations do not require applicants for CPs or OLs under 10 CFR Part 50 to provide PRA-related information. However, the following should be noted:

- The Commission's severe accident policy statement (Ref. 22) articulates the Commission's determination that all new nuclear power plant designs can be shown to be acceptable for severe accident concerns, in part, by completing a PRA and considering the severe accident vulnerabilities the PRA exposes, along with the insights that it may add to the assurance of no undue risk to public health and safety.

- The Commission's advanced reactor policy statement (Ref. 23) articulates that all new nuclear power plant designs should meet the Commission's safety goals (Ref. 24).

- An ongoing rulemaking effort, "Incorporation of Lessons Learned from New Reactor Licensing Process (10 CFR Parts 50 and 52 Licensing Process Alignment)," Docket NRC-2009-0196, RIN-3150-AI66,[3] may add PRA-related requirements for 10 CFR Part 50 CP and OL applications that will be similar to the existing requirements for 10 CFR Part 52 licenses, certifications, and approvals.

On January 14, 2019, the President signed into law the Nuclear Energy Innovation and Modernization Act (Ref. 25). Consistent with Section 103 of the act, the NRC staff has begun to establish a "Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors," Docket NRC-2019-0062, RIN 3150-AK31,[4] for optional use by applicants for new commercial advanced nuclear reactor licenses by December 31, 2027. Specifically, this rulemaking activity will create a new 10 CFR Part 53, which is tentatively titled "Licensing and regulation of advanced nuclear reactors."

3 Further information about this rulemaking (including the proposed schedule) is provided at https://www.nrc.gov/reading-rm/doc-collections/rulemaking-ruleforum/active/ruledetails.html?id=27.

4 Further information about this rulemaking (including the proposed schedule) is provided at https://www.nrc.gov/reading-rm/doc-collections/rulemaking-ruleforum/active/ruledetails.html?id=1108.

The NRC staff positions in this RG applies to NLWRs that are intended to be installed and operated at a fixed site (stationary reactors). These include (1) reactors that are constructed at a site and (2) reactors that are constructed and potentially fueled at an offsite facility and installed at a site. This RG does not address PRAs used to assess the risk during NLWR construction at an offsite facility and transportation to the site.

The staff notes that the regulations in 10 CFR Part 52 requiring DC, SDA, ML, and COL applicants to provide a description of their PRAs and the results, and the regulations in 10 CFR Part 50 requiring COL holders to maintain and upgrade their PRAs, apply to all commercial nuclear power plants, regardless of their design or thermal power. The staff also notes that the Commission's severe accident policy statement and the Commission's advanced reactor policy statement likewise apply to all commercial nuclear power plants. However, in keeping with the philosophy of risk-informed decision-making, the staff recognizes that applicants may want to tailor the PRA's scope and level of detail commensurate with the role that the PRA results play in establishing the licensing basis and regulatory decision-making. Applicants are encouraged to discuss the scope and level of detail that will be provided in their PRAs during pre-application interactions with the NRC staff.

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 IAEA safety requirements and safety guides pursuant to the Commission's "International Policy Statement," published in the *Federal Register* on July 10, 2014 (Ref. 26), and Management Directive and Handbook 6.6, "Regulatory Guides," dated May 2, 2016 (Ref. 27).

The following IAEA safety standards series documents incorporate similar design and pre-operational 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. 28), provides recommendations for meeting the IAEA safety requirements in performing or managing a level 1 probabilistic safety assessment (PSA) project for a nuclear power plant.

- IAEA Safety Standards Series No. SSG-4, "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants," issued 2010 (Ref. 29), provides recommendations for meeting the IAEA safety requirements in performing or managing a level 2 probabilistic safety assessment (PSA) project for a nuclear power plant.

Documents Discussed in Staff Regulatory Guidance

This RG endorses, with exceptions, 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.

C. STAFF REGULATORY GUIDANCE

C.1 An Acceptable Probabilistic Risk Assessment

This section describes one acceptable approach for defining the acceptability of a PRA and its results used in regulatory decision-making for commercial NLWR nuclear power plants. A risk assessment 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., releases of radioactive material, radiological consequences) and their frequencies and (2) is comprised of specific PRA elements for quantifying risk. It is essential that applicants for licenses, certifications, and permits for NLWR designs demonstrate the acceptability of the PRA and its results used to support regulatory decision-making for commercial NLWR nuclear power plants. The same is true for holders of licenses and permits for NLWRs who seek amendments informed by PRA results. The NRC staff assesses the acceptability of the PRA and its results with respect to the scope, level of detail, conformance with national consensus standard PRA elements, and plant representation of a PRA as related to the outcome of the NRC staff's review of a given NLWR licensing application.

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

- Scope of a PRA: The scope of a PRA is defined in terms of (1) the metrics used to characterize risk, (2) the 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 a radioactive release. The scope of a PRA is determined by its intended use for representing the as-built and as-operated plant or the as-designed, as-to-be-built, and as-to-be-operated plant. [5]

- Level of detail of a PRA: The level of detail of a 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 represented. This minimal level of detail is implicit in the elements comprising the PRA and their associated characteristics and attributes.

- Elements of a PRA: The PRA elements are defined in terms of the fundamental technical analyses needed to develop and quantify the PRA model for its intended purpose (e.g., determination of a specific risk metric). The characteristics and attributes of the PRA elements define specific criteria for successfully performing those technical analyses and achieving a defined objective.

- Plant representation and PRA configuration control: Plant representation is defined in terms of how closely the PRA represents the plant as it is designed, built, and operated. In general, PRA results used to support applications after a certificate, approval, permit, or license has been issued should be derived from a PRA model that represents the as-designed, as-to-be-built, or as-to-be-operated plant or as-built, as-operated plant. Consequently, the PRA should be maintained and upgraded, where necessary, to ensure it represents the as-built and as-operated

5 The NLWR PRA standard uses the term "as-intended-to-operate" which is analogous to "as-to-be-operated." "As-to-be-built" refers to the PRA used to model the plant configuration in the preoperational stages of the plant life cycle when 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).

plant through an acceptable configuration control process. Regulatory Position C.1.4 provides guidance on plant representation in the PRA.

C.1.1 Scope of a Probabilistic Risk Assessment

The scope of a PRA used to support an application is defined by the set of initiating events included in the analysis; the set of computed risk metrics; and its intended use for representing the as-built and as-operated plant or the as-designed, as-to-be-built, and as-to-be-operated plant. The process of developing a PRA and its results used to support an application should be complete and comprehensive through consideration of the following:

- All radiological sources at the plant (e.g., reactor cores, spent fuel, fuel reprocessing facilities for molten salt reactors) should be addressed, including accident scenarios that lead to a radioactive release from multiple radiological sources.

- All internal and external hazards should be addressed. For licensing activities, a PRA for the seismic hazard group must always be developed; other hazards should also be included if they cannot be screened out with appropriate justification. Appendix B to this RG lists hazards to consider when developing the PRA.

- All POSs (e.g., at-power and low-power and shutdown (LPSD) types of POSs) should be addressed.

- The frequencies of event sequences should be developed based on the occurrence of an initiating event, evaluation of plant response, evaluation of releases of radioactive material, and the consequences that result from those releases (i.e., an NLWR PRA should address all levels of PRA analysis, analogous to Level 1, 2, and 3 PRAs for LWRs).

Risk characterization for NLWRs is typically expressed by cumulative risk metrics or risk surrogates, commensurate with the purpose for developing the PRA and the role that the PRA plays in regulatory decision-making. The following are two common cumulative risk metrics, which can be directly compared to the quantitative health objectives (QHOs) stated in the Commission’s policy statement on “Safety Goals for the Operation of Nuclear Power Plants”:

- Individual early fatality risk (IEFR): The risk of an early fatality to a biologically average individual (in terms of age and other risk factors) who resides within 1 mile of the site exclusion area boundary. If no individuals reside within 1 mile of the plant boundary, for evaluation purposes, an individual should be assumed to reside 1 mile from the site boundary. An accident may result in the release of a large quantity of radionuclides to the environment that can result in high acute doses to specific organs (e.g., red blood marrow, lungs, lower large intestine) that, in turn, can result in prompt (or early) health effects, fatalities, and injuries. Doses that accumulate during the first week after the accidental release are usually considered when calculating these early health effects. Potential exposure pathways for fatal acute doses typically include inhalation, cloudshine, groundshine, and resuspension inhalation. An early fatality is defined as one that results in death within 1 year of exposure.

- Individual latent cancer fatality risk (ILCFR): The risk of a latent cancer fatality to a biologically average individual who resides within 10 miles of the site. Doses from both acute and chronic exposures, including lifetime 50-year committed doses from early-phase exposure, can result in latent cancer fatalities. These doses arise from exposures that occur during both the early phase (within 1 week of the release) from early-phase exposure pathways such as cloudshine, groundshine, inhalation, and resuspension inhalation, and during the long-term phase from long-term exposure pathways such as groundshine and resuspension inhalation.

Applicants may define risk surrogates subject to the following considerations:

- PRAs of large LWRs use core damage frequency (CDF) and large early release frequency (LERF) as risk surrogates for ILCFR and IEFR, respectively. The definitions of CDF and LERF provided in RG 1.200 may require modification before they can be meaningfully applied to NLWRs, if they can be used at all.

- Large release frequency (LRF) is used as a risk metric for 10 CFR Part 52 DC and COL applications for LWRs, 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. 30). 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. 31), the staff has not developed a definition of LRF. Staff practice has been to allow 10 CFR Part 52 applicants to define LRF.

- In SECY-89-102, "Implementation of Safety Goal Policy," (Ref. 32) the staff proposed a general framework for implementing the Safety Goal Policy Statement that consisted of four principal elements. The fourth principal element addressed the use of subsidiary quantitative targets, which were defined as "…targets that are compatible with but subsidiary to the quantitative safety goal objectives themselves. Subsidiary targets represent a partitioning of safety goal objectives." In SRM-SECY-89-102, "Secy-89-102 – Implementation of the Safety Goals," (Ref. 33) the Commission approved the staff's proposal but noted that "Such subsidiary objectives should be consistent with the large release guideline, and not introduce additional conservatism so as to create a <u>de facto</u> new Large Release Guideline." The Commission further commented that:

    - Within a particular design class (e.g., LWRs, liquid metal reactors (LMRs), high-temperature gas-cooled reactors (HGTRs)), the same subsidiary objectives should apply to both current as well as future designs. A specific subsidiary objective might differ from one specific design class to another specific design class to account for different mitigating concepts (e.g., confinement instead of containment). However, the Large Release Guideline applies to all current as well as all future designs.

- The NLWR PRA standard addresses the development of a PRA that characterizes risk with cumulative risk metrics but also allows the user to define intermediate metrics if justified. Accordingly, the NLWR PRA standard does not provide specific requirements related to the use of risk surrogates.

Each application should define and justify all cumulative risk metrics and risk surrogates used to characterize risk.

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 types of POSs; however, 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 their 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 are a hazard group, whereas a reactor containment building (RCB) breach is a hazard within the internal events hazard group. A hazard group is characterized as either an internal or external hazard type; the distinction between these hazard types is defined by the plant boundary in the PRA. This RG addresses the following seven hazard groups:

- internal events,
- internal flood,
- internal fire,
- seismic events,
- high wind,
- external flood, and
- 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 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. Such a systematic survey of hazards should initially be conducted independent of any pre-determined list of hazards to avoid anchoring bias—the potential for decisions about what hazards should be considered to be influenced by a specific reference point. However, after such an independent survey of hazards is complete, it is reasonable that the results could subsequently be compared to relevant, existing lists of hazards to further assess completeness of the set of hazards to be considered. Thus, a number of internal and external hazards are considered during the development of a 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 way and without the need for a detailed PRA model. Regulatory Position C.1.3.11 provides additional guidance on screening and conservative analyses that can be performed to this end. A hazard that is not categorized under the internal events, internal flood, internal fire, seismic, high wind, or external flood hazards groups is commonly referred to as an "other hazard." Regulatory Position C.1.3.14 provides additional guidance on the modeling of such hazards. An "other hazards" PRA is performed when a screening analysis cannot screen out other hazards. Appendix B to this RG provides a listing and general description of the internal and external hazards that should be considered during the development of a PRA.

Initiating events are perturbations to the steady-state operation of the plant that challenge plant control and safety systems and could lead to plant damage states of interest, 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 Level of Detail of a Probabilistic Risk Assessment

The level of detail of a PRA is defined in terms of the resolution of the modeling used to represent the behavior and operations of the plant. A minimum 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 represented. This minimum level of detail is implicit in the elements making up the PRA and their associated characteristics and attributes.

For a given PRA element and specific PRA analysis elements, the level of detail modeled may vary. The detail may vary from the degree to which (1) plant design and operation are modeled, (2) plant-specific 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 included in the PRA, all technical characteristics and attributes should be addressed. That is, each characteristic and attribute are always addressed, but the degree to which they are addressed may vary.

In general, the level of detail needed in a PRA that supports a risk-informed decision depends on the application under consideration. For reviews of an application, the PRA may be used during different stages of plant design, construction, and operation. Because levels of available information and operating experience vary for each of these stages, the submitted PRA will likewise differ in level of detail for the different stages. For example, a PRA used to support a DC may not have the same level of detail as a PRA for an operational plant that has several years of operating experience. While it is recognized that the level of detail may vary depending on the application, each PRA element and its attributes and characteristics should be addressed.

C.1.3 Elements of a Probabilistic Risk Assessment and Associated Characteristics and Attributes

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

Table 1 lists the PRA elements necessary for an acceptable NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis. A PRA that is missing one or more of these elements would not be considered complete.

Table 1. PRA Elements

• plant operating state analysis	• hazard screening PRA
• initiating event analysis	• high wind PRA
• event sequence analysis	• external flooding PRA
• success criteria development	• other hazards PRA
• systems analysis	• event sequence quantification
• human reliability analysis	• mechanistic source term analysis
• data analysis internal flood PRA	• radiological consequence analysis
• internal fire PRA	• risk integration
• seismic PRA	

These PRA elements are used in the development of an initial PRA model that represents the fundamental plant response to an initiating event, such as equipment or operator failures. A hazard group PRA is developed based on such an initial PRA and also addresses the relationship between the occurrence of a given hazard and the initiating event that starts a given event sequence. For this reason, a hazard group PRA will have some unique analysis requirements that are needed to appropriately represent plant response to a specific hazard in the hazard group. The hazard group-specific PRA analysis elements needed for an acceptable PRA are discussed for each of the hazard group PRA elements. The hazard group-specific PRA analysis elements address the PRA analyses needed specifically for the hazard group under consideration.

The risk integration PRA element is an aspect of PRA acceptability that addresses the integration of all risk contributors from all radiological sources, hazards, POSs, and levels of PRA analysis. The staff did not develop a staff position in RG 1.200 on the risk integration PRA element as it relates to LWR PRA acceptability; however, the staff is promulgating a position in this RG on risk integration as it relates to NLWR PRA acceptability given the scope of the ASME/ANS NLWR PRA standard and to address the needs of regulatory submittals.

C.1.3.1 Plant Operating States Analysis Probabilistic Risk Assessment Element

This section identifies the objectives and the characteristics and attributes of the POS analysis for an NLWR PRA that addresses all radiological sources, all hazards, and all levels of PRA analysis.

The objectives of the POS analysis PRA element are to identify operating evolutions (e.g., full-power, LPSD types of conditions) important to risk and parse them into distinct operating states in which the plant conditions are assumed to be relatively constant. Since the POS analysis PRA element defines the structure of the NLWR PRA, all POSs and the key attributes of the plant conditions in the POSs should be clearly documented in a format (e.g., table or chart) to facilitate understanding of the NLWR PRA results by an independent reviewer. As a plant transitions from design to operation, POS definitions and assumptions used in the PRA for licensing should be reevaluated and modified with as-operated details. The characteristics and attributes needed to achieve the objectives of a POS analysis PRA element are as follows:

- For each plant evolution addressed in the PRA, a set of POSs that represent distinct and relatively constant plant conditions is identified and characterized.
- LPSD plant evolutions are divided into POSs based on differences in plant response to initiating events in a given POS to facilitate the practicality and efficiency of the PRA.
- Each POS to be considered for the specific application is identified and characterized with respect to all important conditions affecting the delineation and evaluation of event sequence families.
- The POS safety functions to consider include reactivity control, reactor coolant chemistry control, decay heat removal control, reactor coolant system (RCS) inventory/barrier control, radionuclide transport barrier control, and ex-vessel fission product control (e.g., off-gas tanks/fuel salt storage tanks/spent fuel pools).
- POS definitions should consider decay heat level, RCS configuration, reactor level (i.e., for reactors with liquid coolant), reactor pressure and temperature, radionuclide transport configuration, status of radionuclide transport barriers, status of fire and flood barriers, available and accurate instrumentation necessary to adequately monitor key plant parameters for the specific POS, and any additional plant parameters and assumed representative plant system configurations needed to determine POS success criteria, POS mechanistic source terms, and POS radiological consequences.
- POS definitions should include all sources of radioactive material within the scope of the PRA, including ex-vessel sources, unless there is a documented technical justification for excluding ex-vessel sources.

- POS definitions should consider any activities that may lead to changes in the above parameters used to define the POS.

- POS definitions should be reviewed to ensure they are adequate for all hazard groups evaluated within the scope of the PRA.

- POS definitions should include consideration of changing plant conditions that may impair or change the effectiveness of hazard barriers, affect propagation pathways, or modify fragilities of SSCs to ensure the appropriateness of the POS definition.

- LPSD types of 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 sources of model uncertainty related to POS definitions and screening are identified and characterized.

- The key attributes of the plant conditions for each POS should be clearly documented in a format (e.g., table or chart) to facilitate understanding of the NLWR PRA results by an independent reviewer.

C.1.3.2 Initiating Event Analysis Probabilistic Risk Assessment Element

This section identifies the objectives and the characteristics and attributes of the initiating event analysis PRA element for an NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis.

The objectives of the initiating event analysis are to identify and characterize events that challenge plant operation during any POSs and that require successful mitigation by plant equipment and personnel to prevent or to mitigate a release of radiological material. Events that have occurred at the plant and those that have a reasonable probability of occurring should be identified and characterized. An understanding of the nature of events will facilitate a grouping of events that allows for managing the many events that can potentially challenge the plant. Initiating event groups should be defined in terms of similar system impacts and plant responses, as based on the related success criteria. The characteristics and attributes needed to achieve the objectives of an initiating event analysis are as follows:

- The analysis includes sufficiently detailed identification and characterization of initiating events.

- Initiating events are grouped so that events in the same group have similar requirements for mitigation.

- Any individual or grouped initiating events are properly screened.

- Initiating event frequency is quantified.

- The sources of model uncertainty related to initiating event analysis PRA element are identified.

- The initiating event analysis PRA element is fully documented to provide traceability of the work.

C.1.3.3 Event Sequence Analysis Probabilistic Risk Assessment Element

This section identifies the objectives and the characteristics and attributes of the event sequence analysis PRA element for an NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis.

The objective of the event sequence analysis PRA element is to model chronologically (to the extent practical) the different possible progressions of events (i.e., event sequences) that can occur from the start of the initiating event to either successful mitigation or release. The event 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 accident progression. The characteristics and attributes needed to achieve the objectives of an event sequence analysis PRA element are as follows:

- The barriers to radionuclide release and the safety functions necessary to protect each barrier for each source are defined in terms of radioactive material sources and described for each POS.

- The analysis should reflect plant-specific dependencies that impact significant event sequences in the event sequence structure.

- The analysis should account for individual function successes, mission times, and time windows for each safety function operator action.

- The analysis should include functional, phenomenological, and operational dependencies and interfaces.

- The analysis should identify and characterize the sources of model uncertainty related to the event sequence analysis PRA element.

- The analysis should fully document the event sequence analysis PRA element to provide traceability of the work.

C.1.3.4 Success Criteria Analysis Probabilistic Risk Assessment Element

This section identifies the objectives and the characteristics and attributes of the success criteria analysis PRA element for an NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis.

The objective of the success criteria analysis PRA element is to determine the minimum requirements for each function (and ultimately the systems used to perform the functions) to prevent 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 depend 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 of interest for prevention of a release and release in each of the reactor-specific release categories, and the computer codes accurately analyze the phenomena of interest. Qualified personnel who are well trained in the use of the codes perform the analyses of interest. The characteristics and attributes needed to achieve the objectives of a success criteria analysis PRA element are as follows:

- Each of the modeled event sequences and event sequence families is defined.

- The analysis defines the key safety functions, supporting systems, structures, radioactive material release barriers, components, and operator actions to support defensible technical basis development.

- The analysis identifies and characterizes the sources of model uncertainty related to the success criteria analysis PRA element.

- The success criteria are fully documented to provide traceability of the work.

C.1.3.5 Systems Analysis Probabilistic Risk Assessment Element

This section identifies the objectives and the characteristics and attributes of the systems analysis PRA element for an NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis.

The objective of the systems analysis PRA element is to identify the various combinations of failures that can prevent a system from performing its function as defined by the success criteria. The model representing the various failure combinations includes the system hardware and instrumentation (and their associated failure modes) and human failure events (HFEs) that would prevent the system from performing its defined functions. 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. The scope of considered HFEs does not include human-induced security events (e.g., sabotage, malevolent acts). The characteristics and attributes needed to achieve the objectives of a systems analysis PRA element are as follows:

- The models are developed in sufficient detail.

- The models reflect the as-designed, as-to-be-built, and as-to-be-operated plant (as applicable).

- The models reflect the success criteria for the systems to mitigate each identified event sequence.

- The models account for both inter- and intra-system dependencies, including support systems, and impacts of dependencies and abnormal environments.

- The models include both active and passive components and failure modes that impact the functions of the system.

- The models include common-cause failures, human errors, unavailability resulting from test and maintenance, and phenomenological effects, as well as dependencies on POSs.

- The models include mission times, failure modes associated with system maintenance, component actuation and functionality, and associated HFEs.

- The models include the sources of model uncertainties related to the system analysis.

C.1.3.6 Human Reliability Analysis Probabilistic Risk Assessment Element

This section identifies the objectives and the characteristics and attributes of the human reliability analysis (HRA) PRA element for an NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis.

The objectives of the HRA PRA element are to identify and define the HFEs that can negatively impact normal or emergency plant operation and quantify their probabilities. 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 or performed incorrectly, 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, considering recovery actions and including any dependencies among actions and conditions.

References such as, but not limited to, NUREG-1792, “Good Practices for Implementing Human Reliability Analysis (HRA),” issued April 2005 (Ref. 34); NUREG-1842, “Evaluation of Human Reliability Analysis Methods Against Good Practices,” issued September 2006 (Ref. 35); and NUREG-2198, “The General Methodology of an Integrated Human Event Analysis System (IDHEAS-G),” issued May 2021 (Ref. 36), provide good practices for meeting the following technical characteristics and attributes that are needed for the HRA PRA element for an internal events PRA during the applicable POSs. The characteristics and attributes needed to achieve the objectives of the HRA PRA element are as follows:

- The HRA PRA element is performed on a POS-by-POS basis.
- The HFEs that would result in initiating events (initiators), and pre- and post-initiator HFEs that would impact the mitigation of initiating events are identified and defined.
- Recovery actions and dependent HFEs are identified.
- The credit for recovery actions is justified.
- Calibration errors or other errors that may impact equipment performance during the applicable POS are considered. (Note: The calibration errors or other errors may occur at a different POS from the POS being analyzed.)
- The associated human error probabilities are quantified considering scenario- and plant-specific factors and including appropriate dependencies (e.g., between pre- and post-initiator HFEs).
- The sources of model uncertainty related to the HRA PRA element are identified and analyzed.

C.1.3.7 Data Analysis Probabilistic Risk Assessment Element

This section identifies the objectives and the characteristics and attributes of the data analysis PRA element for an NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis. The objectives of the data analysis PRA element are as follows:

- Clearly define the parameter boundaries.
- Appropriately group components.

- Ensure that the parameter data are consistent with parameter definitions.

- Include relevant generic industry, design-specific, and plant-specific evidence in the parameter estimation.

- Address uncertainty parameters.

- Fully document the data analysis PRA element to provide traceability of the work.

The data analysis PRA element quantifies the equipment failure probabilities and equipment unavailabilities of the modeled systems. The estimation process includes a mechanism for addressing uncertainties and has an ability to combine different sources of data in a coherent manner, including the relevant generic information and actual operating history and experience of the plant when it is of sufficient quality, as well as applicable generic experience. The characteristics and attributes needed to achieve the objectives of a data analysis PRA element are as follows:

- The estimation of parameters associated with basic event probability models and unavailability events uses generic, design-specific, plant-specific data, or a combination of the three as applicable. Each parameter is clearly defined in terms of the logic model and the model used to evaluate event probability.

- Estimation is based on the relevant generic industry and technology- and design-/plant-specific evidence.

- Estimation considers the design, environmental, and service conditions of the components in the as-designed, as-to-be-built, and as-to-be-operated plant.

- Estimation is consistent with component boundaries.

- Estimation includes identification and characterization of the uncertainty.

C.1.3.8 Internal Flood Probabilistic Risk Assessment Element

This section identifies the hazard group-specific PRA analysis elements, the objectives of those analysis elements, and the characteristics and attributes that are needed for an acceptable internal flood NLWR PRA that addresses all radiological sources, all POSs, and all levels of PRA analysis.

The objectives of each internal flood-specific PRA analysis element are briefly described and the characteristics and attributes needed to achieve the objective are provided below. The internal flood-specific PRA analysis elements are evaluated for all POSs and may have different characteristics across POSs. The following internal flood-specific PRA analysis elements are applicable to all phases leading up to and including the as-built, as-operated plant:

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

PRA models of internal floods are based on an internal events 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 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.1.3.1 through C.1.3.7 of this RG, as necessary.

The internal flood PRA for at-power and LPSD types of POSs are similar in many ways, differing primarily in plant configuration, including radioactive or hazardous material inventory distribution, or both, and temporary features. These differences can manifest themselves in the flood pathways and water levels; internal flood-induced failure probability of structures, systems, and components (SSCs); the plant response; or a combination of these three as compared to at-power-types of POSs.

The objective of **internal flood area partitioning** is to divide the plant into flood areas that are used as the basis for the analysis. Flood areas are defined on the basis of physical barriers, mitigation features, and propagation pathways. All POSs should be evaluated for differences in the internal flood area partitioning analysis element. The differences in the POSs may impact the flood areas that are used. The characteristics and attributes needed to achieve the objectives of an internal flood area partitioning are as follows:

- Flood areas are defined based on plant features that can restrict a flood.

- Area definitions are verified through plant walkdowns or by evaluating available data and findings of investigations of the plant design and operations information for plants that have not started construction or do not have enough construction complete to allow physical walkdowns.

- Flood areas are based on the physical barriers, mitigation features, and propagation pathways for all POSs.

- Sources of model uncertainty are identified and characterized for plant partitioning.

The objective of the **internal flood source analysis** is to identify 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. All POSs should be evaluated for differences in the internal flood source analysis element. It is important that the differences in POSs are considered in evaluating flood sources. The characteristics and attributes needed to achieve the objectives of the internal flood source analysis are as follows:

- The identification and characterization of the following are sufficiently detailed:

  - SSCs located within each area,
  - flood sources and flood mechanisms for all POSs, and
  - type of water release and capacity.

- Well defined and justified screening criteria are used for the elimination of flood sources and areas. Information is verified through plant walkdowns or by evaluating available data and findings of investigations of the plant design and operations information for plants that have not started construction or do not have enough construction complete to allow physical walkdowns.

The objective of the **internal flood scenario analysis** is to identify 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) for all POSs. 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 (e.g., submergence, spray, pipe whip, and jet impingement) is examined. 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. All POSs should be evaluated for differences in the internal flood scenario analysis element. It is important that the flood sources and flood propagation paths consider the possible differences between POSs and possible temporary alignments that could impact the evaluation. These differences from at-power configurations may account for changes in flood pathways, in flood barrier locations and capabilities, in the location of SSCs, and additions of temporary features. The characteristics and attributes needed to achieve the objectives of the internal flood scenario analysis are as follows:

- The following are identified and evaluated:
  - flood propagation paths for all POSs,
  - flood-mitigating plant design features (e.g., drains and sumps) and operator actions, and
  - the susceptibility of SSCs in each flood area to the different types of floods.
- Well defined and justified screening criteria are used for the elimination of flood scenarios.
- All POSs should be evaluated in the internal flood scenario analysis element.
- Differences in the POS plant configuration and propagation pathways should be considered.

The objective of the **internal flood scenario delineation and quantification** is to provide an estimation of the source terms and the radiological consequences of the plant that includes internal floods. The frequency of flood-induced initiating events that represent the design, operation, and experience of the plant is quantified. The internal events PRA is modified and the internal flood event sequences are quantified to (1) modify event 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 HRA to account for performance-shaping factors that are attributable to flooding, and (5) quantify internal flood source terms and radiological consequences. All POSs should be evaluated for differences in the internal flood scenario delineation and quantification analysis element. The characteristics and attributes needed to achieve the objectives of the internal flood scenario delineation and quantification are as follows:

- Flood-induced initiating events are identified and grouped on the basis of a structured and systematic process for all POSs.
- Flood-initiating event frequencies are estimated.
- The internal events PRA is modified to account for flooding effects, including uncertainties.
- The uncertainties in the internal flood PRA for a POS are characterized. The potential impact of sources of model uncertainty and related assumptions on the results are justified.
- Source terms and radiological consequences for chosen flood sequences are estimated.
- Well defined and justified screening criteria are used for the elimination of flood scenarios.

C.1.3.9 Internal Fire Probabilistic Risk Assessment Element

This section identifies the hazard group-specific PRA analysis elements, the objectives of those analysis elements, and the characteristics and attributes that are needed for an acceptable internal fire NLWR PRA that addresses all radiological sources, all POSs, and all levels of PRA analysis.

The objective for each technical hazard group-specific PRA analysis element is briefly described, and the characteristics and attributes needed to achieve the objective are given below. The internal fire-specific PRA analysis elements are evaluated for all POSs and may have different characteristics for different POSs. The internal fire-specific PRA analysis elements for an internal fire PRA at all phases leading up to and including the as-built, as-operated plant are as follows:

- internal fire plant boundary definition and partitioning,
- internal fire initiating event and equipment selection,
- internal fire cable selection and location,
- internal fire qualitative screening,
- internal fire plant response model,
- internal fire scenario selection and analysis,
- internal fire ignition frequency,
- internal fire circuit failure analysis,
- internal fire HRA, and
- internal fire event sequence quantification.

Internal fire PRA models for at-power and LPSD types of POSs are similar in many ways, differing primarily in the relevant operating experience and plant configuration. The internal fire PRA model for a particular POS also relies on the corresponding internal events POS PRA model, which is modified to reflect fire-induced failure of equipment causing initiating events, to reflect fire-induced failure of equipment used to respond to initiating events, and to reflect the impact of fire on operator actions. Because of its dependence on the internal events model, the internal fire analysis incorporates the elements of Regulatory Positions C.1.3.1 through C.1.3.7 of this RG, as necessary.

The objective of the **internal fire plant boundary definition and partitioning** is to establish the overall boundaries of the fire PRA and divide the area within that boundary into smaller regions (i.e., physical analysis units (PAUs)), commonly known as fire areas or compartments. The entire fire PRA is generally organized according to these PAUs. The at-power boundary definition and partition for an at-power type of POS may need to be modified for boundary elements breached during LPSD types of POSs, but not breached during at-power types of POSs. The plant boundary definition and partitioning should account for PAUs necessary for all POSs. The characteristics and attributes needed to achieve the objectives of an internal fire plant boundary definition and partitioning PRA analysis element are as follows:

- The global analysis boundary should account for all plant locations relevant to the internal fire PRA for all POSs.

- PAUs are identified by credited partitioning elements that can substantially confine fire damage behaviors.

- The boundary definition and partition for at-power types of POSs may need to be modified for boundary elements breached during LPSD types of POSs but not breached during at-power types of POSs.

- The plant boundary definition and partitioning should account for the PAUs necessary for all POSs.

- The uncertainties in the internal fire PRA related to the PAUs are identified and characterized.

The objective of the **internal fire initiating event and equipment selection** is, for each POS, to identify the internal fire-induced initiating events to be evaluated in the fire PRA model and equipment to be included in the internal fire PRA model. Much of this equipment comes from the equipment included in the internal events PRA such that, if failed by an internal fire, that equipment could produce a plant initiator or affect the plant response. The plant's fire protection program and analysis can be used to identify equipment. The critical safety functions essential to the LPSD model are reactivity control, reactor coolant chemistry control (key for NLWRs), decay heat removal control, RCS inventory/barrier control, and ex-vessel fission product control (e.g., off-gassing/fuel salt storage tanks). Internal fire-induced spurious actuations are of particular interest for initiating event and equipment selection. The selected equipment is mapped to the PAUs. The internal fire PRA model for each POS should be evaluated for the need for different or additional equipment, particularly in the case of spurious actuations. The characteristics and attributes needed to achieve the objectives of an internal fire-initiating event and equipment selection PRA analysis element are as follows:

- Fire-induced initiating events to be evaluated in the internal fire PRA model are identified.

- Equipment is included in the internal fire plant response model that will lead 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 their consequence (e.g., internal fire-induced failures leading to loss of heat sink or radionuclide transport barrier bypass require a greater number of spurious operations to be included in the fire plant response model).

- Instrumentation and support equipment are included.

- The internal fire PRA model for each POS should be evaluated for the need for different or additional equipment, particularly in the case of spurious actuations.

- The uncertainties in the internal fire PRA related to the internal fire-initiating events and equipment selection are identified and characterized.

The objective of the **internal fire cable selection and location** is to identify those cables associated with the equipment identified in the internal fire-initiating event and equipment selection technical element. The selected cables are mapped to the PAUs and, in some cases, to electrical raceways. The ability to locate a cable for the internal fire PRA is limited by the information known about the plant (i.e., the lack of as-built details) prior to construction. The location of cables is not generally affected by the particular POS, unless cables are temporarily routed during that POS. The characteristics and attributes needed to achieve the objectives of an internal fire cable selection and location PRA analysis element are as follows:

- Cables that are required to support the operation of equipment represented in the internal fire PRA (defined in the equipment selection element) are identified and located.

- The ability to locate a cable for the internal fire PRA is limited by the information known about the plant (i.e., the lack of as-built details) prior to construction.

- The location of cables is not generally affected by the POS, unless cables are temporarily routed during the POS.

- The uncertainties in the internal fire PRA related to cable selection are identified and characterized.

The objective of the **internal fire qualitative screening** is to eliminate certain PAUs defined in the plant boundary definition and partitioning element that can be shown to be unimportant to fire risk for a POS. These screening criteria should be general qualitative criteria. Those PAUs screened out in the internal fire qualitative screening PRA analysis element play no role in the more detailed quantitative assessment. The characteristics and attributes needed to achieve the objectives of an internal fire qualitative screening PRA analysis element are as follows:

- Qualitatively screened out PAUs represent negligible contributions to risk and are considered no further for a POS.

- The uncertainties in the internal fire PRA associated with qualitative screening are identified and characterized.

The objective of the **internal fire plant response model** is to develop a logic model that represents the plant response following an internal fire. This model is based on the internal events PRA model for a POS. The internal events PRA model for a POS is modified to account for fire effects, including modifications due to SSC failures that specifically result from fire and consideration of fire-specific procedures. The latter are processed through the internal fire human reliability PRA analysis element. The characteristics and attributes needed to achieve the objectives of an internal fire plant response model PRA analysis element are as follows:

- Based on the internal events PRA, the logic model for a POS is adjusted to add new internal fire-induced initiating events and modified or new event sequences, operator actions, and accident progressions (in particular those from spurious actuations).

- Issues relevant to the internal fire PRA (e.g., those relevant findings from a peer review of the internal events PRA) are resolved and incorporated into the fire plant response model.

- Inapplicable aspects of the internal events PRA model are bypassed for a particular POS.

- The uncertainties in the internal fire PRA related to the internal plant response model are identified and characterized.

The objective of the **internal fire scenario selection and analysis** is to define and analyze fire event scenarios that represent the plant fire risk associated with each PAU. Internal 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 internal fire. Main control room internal fire scenarios, including control room abandonment, are analyzed explicitly. Multicompartment fire propagation scenarios, including scenarios from all screened PAUs, are also assessed and screened as appropriate. The ability to develop internal fire scenarios in the fire PRA is limited by the information known about the plant (i.e., the lack of as-built details) prior to construction. Particularly for LPSD types of POSs, data is important in establishing the availability of fire protection features and systems, including the status of those fire protection features and systems during the particular POS and the plant conditions under which they are available. Also, the nature and amount of transient fuel sources introduced in the plant during the POS may differ between at-power and LPSD types of POSs. All POSs should be evaluated for differences in the internal fire scenario selection and analysis PRA analysis element. The characteristics and attributes needed to achieve the objectives of an internal fire scenario selection and analysis PRA analysis element are as follows:

- Internal 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 internal fire modeling tools are applied.
- The technical basis is established for statistical and empirical models in the context of the internal fire scenarios (e.g., fire brigade response).
- Scenarios involving the internal fire-induced failure of structural steel are identified and assessed (at least qualitatively).
- Multicompartment fire propagation scenarios are also assessed and screened as appropriate.
- The ability to develop internal fire scenarios in the internal fire PRA is limited by the information known about the plant (i.e., the lack of as-built details) prior to construction.
- The nature and amount of transient fuel sources introduced in the plant may differ for different POSs and should be evaluated.
- The availability of fire protection features and systems during a POS should be evaluated for plant activities that have a bearing on their availability. For example, dependencies between activities in the plant, such as removing a fixed suppression system from service in an area while performing hot work, should be addressed.
- Fire barrier failures should also be addressed in the context of those plant activities leading to the demand for that barrier.
- All POSs should be evaluated for differences in the internal fire scenario selection and analysis sub-element.
- The uncertainties in the internal fire scenario selection and analysis are identified and characterized.

The objective of the **internal fire ignition frequency** is to estimate the frequencies of the ignition sources postulated for the internal 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 internal fire event frequencies, possibly augmented with plant-specific experience for operating reactors, 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. Internal fire ignition frequencies due to LPSD types of POS conditions that are different from conditions in at-power types of POSs should be addressed (e.g., the frequency of general transient fires or hot work fires).

All POSs should be evaluated for differences in the internal fire ignition frequency sub-element. The characteristics and attributes needed to achieve the objectives of an internal fire ignition frequency PRA analysis element are as follows:

- Frequencies are established for ignition sources and consequently for PAUs.

- Transient fires should be postulated for all PAUs regardless of administrative controls.

- Appropriate justification should be provided for the use of nonnuclear experience to determine internal fire ignition frequency.

- Internal fire frequencies should be specific to POS conditions (e.g., the frequency of general transient fires or hot work fires) as appropriate.

- All POSs should be evaluated for differences in the internal fire ignition frequency PRA analysis element.

- The uncertainties related to the internal fire ignition frequencies are identified and characterized.

The objective of the **internal fire circuit failure analysis** is to treat the impact of internal fire-induced circuit failures on the plant response for all POSs. In particular, spurious actuations from hot shorts are analyzed. The conditional probability of the particular circuit failure is identified and assigned. The characteristics and attributes needed to achieve the objectives of an internal fire circuit failure analysis PRA analysis element are as follows:

- The conditional probability of occurrence of various circuit failure modes given cable damage from an internal fire is based on cable and circuit features.

- The ability to develop internal fire-induced circuit failure likelihoods in the internal fire PRA is limited by the information known about the plant (i.e., the lack of as-built details) prior to construction.

- Since the cable itself, its function in the plant, and cable location relative to other cables are not often affected by the LPSD type of POSs, the circuit failure analysis from at-power types of POSs is often applicable to the LPSD type of POSs, with limited potential exceptions. Differences in circuit failure analysis should be evaluated for different POSs.

- The uncertainties related to the internal fire circuit failure analysis are identified and characterized.

The objective of the **internal fire HRA** is to identify operator actions and related HFEs, both within and outside the main control room, for inclusion in the plant response model for the POS. This element also includes quantification of human error probabilities for the modeled actions. Modeled operator actions include those introduced into the plant response model resulting strictly from internal fire-related procedures and those actions retained from the internal events PRA. The latter HFEs are modified to account for internal fire effects. The characteristics and attributes needed to achieve the objectives of an internal fire HRA PRA analysis element are as follows:

- Operator actions and related post-initiator HFEs, conducted both within and outside of the main control room, are addressed.

- The effects of internal fire-specific procedures are identified and incorporated into the plant response model.

- Plausible and feasible recovery actions, assessed for the effects of internal 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 internal fire PRA for a particular POS are assessed for fire effects.

- The uncertainties related to the internal fire HRA PRA analysis element are identified and characterized.

The objective of the **internal fire event sequence quantification** is to calculate the frequency of the internal fire-induced event sequence (i.e., the fire ignition frequency and the probability of fire damage). This factor is then integrated with the conditional probability of the event sequence from the internal fire PRA plant response model for the appropriate POS to quantify the risk. In the internal fire event sequence quantification PRA analysis element, dependencies are addressed, risk-significant contributors to event sequences are identified, the uncertainty in PRA results is characterized, and the event sequence quantification results are reviewed for correctness, completeness, and consistency. The characteristics and attributes needed to achieve the objectives of an internal fire sequence quantification PRA analysis element are as follows:

- For each internal fire scenario, the internal fire risk results are quantified by combining the internal fire ignition frequency, the probability of fire damage, and the conditional probability of the event sequence from the internal fire PRA plant response model for a particular POS.

- Total risk is calculated for the plant, and risk-significant contributors to event sequences are identified.

- Identified dependencies are addressed.

- Uncertainties in the internal fire PRA for a POS are characterized. The potential impact of sources of model uncertainty and related assumptions on the results are justified.

- Internal fire scenarios may be screened out in the internal fire event sequence quantification PRA analysis element based on preestablished screening criteria for all POSs.

C.1.3.10 Seismic Probabilistic Risk Assessment Element

This section identifies the hazard group-specific PRA analysis elements, the objectives of those analysis elements, and the characteristics and attributes that are needed for an acceptable seismic NLWR PRA that addresses all radiological sources, all POSs, and all levels of PRA analysis.

The objective of each seismic PRA analysis element is briefly described, and the characteristics and attributes needed to achieve the objective are provided. It is assumed that the seismic PRA for a given POS is based on modifications made to a corresponding up-to-date internal events PRA. The seismic PRA analysis element is evaluated for all POSs and may have different characteristics across POSs. The following are the seismic PRA analysis elements, applicable to all phases leading up to and including the as-built, as-operated plant:

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

Earthquakes can cause initiating events different from 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. A probabilistic seismic analysis considers all possible levels of earthquakes, along with their frequencies of occurrence and consequential damage to plant systems and components. Because of its dependence on the internal events model, the seismic PRA incorporates the elements of Regulatory Positions C.1.3.1 through C.1.3.7 of this RG, as necessary.

The seismic PRA development for at-power and LPSD types of POSs are similar in many ways, differing primarily in plant configuration, including radioactive or hazardous material inventory distribution, or both, and temporary features. These differences can manifest themselves in the seismic capacity of SSCs, the plant response, or both as compared to at-power-types of POSs.

The objective of the **seismic hazard analysis** is 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. When 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 the site-specific probabilistic seismic hazard.

In a probabilistic seismic hazard analysis, an essential part of the methodology is the consideration of uncertainties associated with the randomness of events and the state of knowledge (i.e., aleatory and epistemic uncertainties). This typically results in the generation of a set of hazard curves, defined at specified fractile (confidence) levels and a mean hazard curve. It is likely that a specific site would not be identified during the design phase. In such a case, a representative or bounding site can be identified with justification that the site is either representative of or bounding for the anticipated sites for the reactor, and the seismic hazard analysis PRA analysis element discussed above should be applied to that representative or bounding site. Various plant POSs are expected to be evaluated using the same seismic hazard analysis. The characteristics and attributes needed to achieve the objectives of a seismic hazard analysis element are as follows:

- The frequency of ground motion levels at a site is established.

- A specific site is identified, or a representative or bounding site is identified with justification.

- All credible sources of damaging earthquakes are examined.

- Information is current.

- The analysis is based on comprehensive data, including geological, seismological, and geophysical data, local site topography and historical information.

- The analysis reflects the composite distribution of the informed technical community.
- The level of analysis depends on the application and site complexity.
- The hazard analysis considers uncertainties in characterizing the seismic sources and the ground motion propagation. Uncertainties should—
  - be properly accounted for,
  - be fully propagated,
  - allow estimates of fractile hazard curves and median and mean hazard curves, and
  - reflect uniform hazard response spectra.
- Features of spectral shapes used in the seismic PRA include the following:
  - They should be 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 they reflect the site-specific shape.
- The analysis should assess whether other seismic hazards should be included in the seismic PRA, such as fault displacement, landslide, soil liquefaction, or soil settlement.

The objective of the **seismic fragility analysis** is to estimate 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 for various POSs, as confirmed through a detailed walkdown of the plant. The fragilities of all the systems modeled in the event sequences for each POS are included. It is likely that a specific site would not be identified during the design phase. In addition, the actual current configuration of SSCs in the plant and its confirmation by a detailed physical walkdown of the plant may not be feasible during the design and construction phases. In such cases, assumptions used in seismic fragility analysis (e.g., seismic motion parameters, SSC configuration and design characteristics) should be clearly identified, documented, and tracked to ensure their continued validity across different stages. Such assumptions include those that are identified or included in virtual layouts of the plant. All POSs should be evaluated for differences in the seismic fragility analysis PRA analysis element. It is important that the walkdowns evaluate the differences between the different POSs that impact the fragility analysis. The fragility analysis may need to be modified for LPSD types of POSs to account for changes compared to configurations for at-power types of POSs including but not limited to changes in the location of SSCs, in the radioactive or hazardous material inventory in SSCs, or both, and the addition of temporary features. The characteristics and attributes needed to achieve the objectives of a seismic fragility analysis PRA analysis element are as follows:

- The seismic fragility estimate—
  - is plant-specific,
  - is realistic,

- includes all SSCs that are involved in event sequences for each POS modeled in the seismic PRA systems model, and
- describes the basis for screening of high-capacity components.

- Seismic fragility evaluation is performed for SSCs for each POS based on the following:
  - review of plant design documents,
  - plant configuration,
  - earthquake experience data,
  - fragility test data,
  - generic qualification test data (with justification),
  - analytical approaches using plant- and location-specific seismic demand information, and
  - walkdowns or the evaluation of available data and findings of investigations of the plant design and operations information for plants that have not started construction or do not have enough construction complete to allow physical walkdowns.
- Plant walkdowns (or the evaluation of available data and findings of investigations of the plant design and operations information for plants that have not started construction or do not have enough construction complete to perform physical walkdowns) are performed for all applicable POSs and include but are not limited to walkdowns of anchorage, lateral seismic support and potential seismic system interactions.
- Uncertainties related to the seismic fragility are identified and characterized.

The objective of the **seismic plant response analysis** is to determine the plant response to and radiological consequences of a seismic event for each POS by combining the plant logic model for each POS with the corresponding component fragilities and the seismic hazard estimates. Usually, the analysis is based on the internal events PRA model for each POS. Unique aspects of the seismic event are incorporated by adding basic events for seismic-induced failures for each POS to the corresponding internal events PRA model. Some portions of the internal events PRA model for a POS that do not apply or that can be screened out based on the impact on the base seismic PRA should be eliminated. For example, near-term 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. The seismic PRA model for each POS includes all applicable significant seismic causes, initiating events, and seismic-induced SSC failures, as well as significant non-seismic failures and human errors. All POSs should be evaluated for differences in the seismic plant response analysis PRA analysis element. It is important that the walkdowns evaluate the differences between the different POSs that impact the plant response analysis. The characteristics and attributes needed to achieve the objectives of a seismic plant response analysis PRA analysis element are as follows:

- The seismic PRA model for all POSs includes the following:
  - seismic-caused initiating events,

- seismic-induced SSC failures,
- non-seismic-induced unavailabilities, and
- other significant failures (including human errors) that can contribute to seismic risk and radiological consequences.

- The seismic PRA model is adapted to incorporate seismic analysis aspects that are different from the corresponding internal events PRA model.

- The seismic PRA model reflects the as-designed or as-to-be built and as-to-be operated or as-built and as-operated plant being analyzed.

- Quantification of risk metrics for each POS integrates the following:
  - the seismic hazard analysis,
  - the seismic fragilities analysis, and
  - the plant response logic analysis.

- The uncertainties related to the seismic plant response model are identified and characterized.

The seismic PRA model reflects the as-built and as-operated plant or the as-designed or the as-to-be built and as-to-be operated plant, as applicable in each stage. Assumptions used in seismic plant response analysis (e.g., screening out seismic-induced failures, human error event identification and development) should be clearly identified, documented, and tracked to ensure their continued validity across different stages.

In meeting the technical characteristics and attributes for the seismic portion of an external hazard PRA, a seismic margins method is outside the scope of this RG and would be addressed on a case-by-case basis.

C.1.3.11 Hazards Screening Analysis Probabilistic Risk Assessment Element

This section identifies the objectives and the characteristics and attributes for a hazards screening analysis for an NLWR PRA that addresses all radiological sources, all POSs, and all levels of PRA analysis.

The objective of the hazards screening analysis is to systematically identify all natural and human-caused hazards and, if screening is performed, to adequately justify exclusion of a hazard or hazard group. Screening methods can often be used to show that the contribution of a hazard or hazard group to a risk metric (e.g., radiological doses, health effects to the public) is not significant. Table B-1 in Appendix B to this RG lists the hazards that should be addressed in the PRA. However, to help ensure that analysis resources can be applied to the more important contributors to risk, in many cases, some of the hazards or hazard groups in Table B-1 may be excluded from a detailed PRA if they can be shown to meet predefined and justified screening criteria. For hazards or hazard groups that are screened out from further consideration in a PRA, the justification for screening them out should be archived and may need to be reevaluated in subsequent evaluations, depending on the application, to confirm that the screening remains appropriate.

A preliminary screening analysis may be performed to demonstrate that pre-defined qualitative or semi-quantitative screening criteria are met and the hazard or hazard group under consideration can be excluded from further analysis in the PRA. Preliminary screening analyses generally involve more simple and less involved analysis (e.g., demonstrating a hazard or hazard group is not physically realizable at a site or range of sites). Detailed screening analyses may be performed to demonstrate that pre-defined quantitative screening criteria are met and the hazard or hazard group under consideration can be excluded from further analysis in the PRA. Detailed screening analyses should be performed using a bounding or demonstrably conservative analysis to develop a quantitative estimate of risk. Walkdowns of the plant site and plant buildings or the evaluation of available data and findings of investigations of the plant design and operations information are used to confirm assumptions that help form the basis for screening. For some applications, the staff may need to examine and confirm the validity of the assumptions used to screen out a hazard from the PRA by using application-specific guidance. The characteristics and attributes needed to achieve the objectives of a hazards screening analysis are as follows:

- All potential hazards that can affect the design, plant, or site or that are unique to a specific design, plant, or site are systematically identified.

- A preliminary screening is performed using predefined qualitative screening criteria.

- A detailed screening analysis is performed using predefined quantitative screening criteria and should involve bounding or conservative analyses.

- The basis for any screening analysis is confirmed with a walkdown for plants with a selected site or by evaluating available data and findings of investigations of the plant design and operations information for plants without a selected site.

- The uncertainties related to hazard screening are identified and characterized.

C.1.3.12 High Wind Probabilistic Risk Assessment Element

This section identifies the hazard group-specific PRA analysis elements, the objectives of those analysis elements, and the characteristics and attributes that are needed for an acceptable high wind NLWR PRA that addresses all radiological sources, all POSs, and all levels of PRA analysis.

The objective of each high wind PRA analysis element is briefly described, and the characteristics and attributes needed to achieve the objective are provided below. The high wind PRA analysis elements for all POSs are as follows:

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

The types of high wind events that should be considered in the analysis are site dependent. They can include tornadoes, tropical cyclones (i.e., hurricanes, typhoons), thunderstorms, squall lines, and other weather fronts that produce high winds. 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.1.3.1 through C.1.3.7 of this RG, as necessary.

The objective of a **high wind hazard analysis** is to estimate 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 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. The characteristics and attributes needed to achieve the objectives of the high wind hazard analysis PRA analysis element are as follows:

- A probabilistic wind hazard analysis—
  - results in frequency of high wind at the site,
  - is based on site-specific data, and
  - reflects recent information.

- Uncertainties in the models and parameter values—
  - are properly accounted for,
  - are fully propagated, and
  - allow estimate of the mean hazard curve.

The objective of a **high wind fragility analysis** is to estimate plant-specific, realistic wind fragilities for those SSCs (or their combination) whose failure contributes to plant risk or radiological consequences. The characteristics and attributes needed to achieve the objectives of high wind fragility analysis are as follows:

- The analysis is plant-specific.

- The analysis is realistic.

- All SSCs whose failure contributes to core damage or large early release are included.

- The analysis is confirmed through plant walkdowns or by evaluating available data and findings of investigations of the plant design and operations information for plants that have not started construction or do not have enough construction completed to allow physical walkdowns.

- The uncertainties related to the high wind fragility analysis are identified and characterized.

The objective of a **high wind plant response analysis** is to develop a high wind PRA systems model that includes all significant high-wind-induced initiating events and other failures that can lead to plant risk or radiological consequences. The model is adapted from the internal events model to incorporate unique high wind analysis aspects that are different from those in the internal events PRA model. The characteristics and attributes needed to achieve the objectives of high wind plant response analysis PRA analysis element are as follows:

- All significant high-wind-induced initiating events are included.

- Other significant failures (both those that are wind induced and those that are random failures) that can lead to consequence metrics are included.

- The high wind PRA systems model is adapted from the internal events PRA model for all modeled POSs and radiological sources.

- The model incorporates unique high wind analysis aspects that are different from those in the internal events PRA model.

- The uncertainties related to high wind plant response analysis PRA analysis element are identified and characterized.

C.1.3.13 External Flooding Probabilistic Risk Assessment Element

This section identifies the hazard group-specific PRA analysis elements, the objectives of those analysis elements, and the characteristics and attributes that are needed for an acceptable external flood NLWR PRA that addresses all radiological sources, all POSs, and all levels of PRA analysis.

The objective for each technical element is briefly described, and the characteristics and attributes needed to achieve the objective are provided below. It is assumed that the external flood PRA for a POS is based on modifications made to a corresponding up-to-date internal events PRA. The technical elements are evaluated for all POSs and may have different characteristics across POSs. The technical elements for an external flood PRA, applicable to all phases leading up to and including the as-built, as-operated plant, are as follows:

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

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, are considered. Because of its dependence on the internal events model, the external flood PRA incorporates the elements of Regulatory Positions C.1.3.1 through C.1.3.7 of this RG, as necessary.

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 types of PRA analysis elements for an external flood PRA are similar to those for an internal flood PRA and a seismic PRA.

The external flood PRAs for at-power and LPSD types of POSs are similar in many ways, differing primarily in plant configuration, including radioactive or hazardous material inventory distribution, or both, and temporary features. These differences can manifest themselves in the flood pathways and water levels, external flood-induced failure probability of SSCs, or the plant response for the LPSD types of POSs as compared to at-power types of POSs.

The objective of an **external flood hazard analysis** is to estimate the frequency of external floods 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. It is likely that a specific site would not be identified during the design phase. In such a case, a representative or bounding site can be identified with justification that the site is either representative of or bounding for the anticipated sites for the reactor, and the seismic hazard analysis discussed above should be applied to that representative or bounding site. Various plant POSs are expected to be evaluated using the same external flood hazard analysis. The characteristics and attributes needed to achieve the objectives of an external flood hazard analysis PRA analysis element are as follows:

- Probabilistic flood hazard analysis—
  - results in frequency of external floods at the site,
  - is based on site-specific data or data for a justified representative or bounding site, as applicable, and
  - reflects recent information.
- Uncertainties in the models and parameter values—
  - are properly accounted for,
  - are fully propagated, and
  - allow estimation of the mean hazard curve.

The objective of an **external flood fragility analysis** is to perform an evaluation to estimate plant-specific, realistic flood fragilities for those SSCs (or their combination) in each POS whose failure contributes to risk from an external flood hazard. It is likely that a specific site would not be identified during the design phase. In addition, actual current configuration of SSCs in the plant and its confirmation by a detailed physical walkdown of the plant may not be feasible during the design and construction phases. In such cases, assumptions used in the external flood fragility analysis PRA analysis element (e.g., location of flood barriers, SSC configuration and design characteristics) should be clearly identified, documented, and tracked to ensure their continued validity across different stages. Such assumptions include those that are identified or included in virtual layouts of the plant. All POSs should be evaluated for differences in the external flood fragility analysis PRA analysis element. It is important that the walkdowns evaluate the differences between the different POSs that impact the fragility analysis. The fragility analysis may need to be modified for LPSD types of POSs to account for changes compared to configurations for at-power types of POSs including but not limited to changes in flood pathways, in the location of SSCs, in the radioactive or hazardous material inventory in SSCs, or both, and addition of temporary features. The characteristics and attributes needed to achieve the objectives of an external flood fragility analysis are as follows:

- The flood fragility estimate—
  - is plant specific,
  - is realistic, and
  - includes all SSCs that are involved in event sequences for each POS modeled in the external flooding PRA systems model.
- An external flooding fragility evaluation is performed for SSCs for each POS based on the following:
  - a review of plant design documents,
  - plant configuration, and

- a walkdown or the evaluation of available data and findings of investigations of the plant design and operations information for plants that have not started construction or do not have enough construction complete to allow physical walkdowns.

- The uncertainties related to external flood fragility analysis are identified and characterized.

The objective of an **external flood plant response analysis** is to develop an external flood PRA model that includes all significant flood-caused initiating events and other failures that can contribute to the plant response to and radiological consequences from external flooding events for each POS. The model for each POS is adapted from the corresponding internal events PRA model to incorporate unique flood analysis aspects that are different from the internal events PRA model. The external flooding PRA model for each POS includes all applicable significant external flooding causes, initiating events, and external flooding-induced SSC failures, as well as significant non-seismic failures and human errors. All POSs should be evaluated for differences in the external flooding plant response analysis PRA analysis element. It is important that the walkdowns evaluate the differences between the different POSs that impact the plant response analysis. The characteristics and attributes needed to achieve the objectives of an external flood plant response analysis PRA analysis element are as follows:

- The external flood PRA model for all POSs—
    - includes all significant external flood-caused initiating events,
    - includes other significant failures (both those that are caused by the flood and those that are random failures) that contribute to external flooding risk and radiological consequences,
    - is adapted from the internal events PRA model for all modeled POSs and radiological sources,
    - incorporates unique external flood analysis aspects that are different from the internal events PRA model, and
    - reflects the as-designed or as-to-be built and as-to-be operated or as-built and as-operated plant being analyzed.
- Quantification of risk metrics for each POS integrates the following:
    - the external flooding hazard analysis,
    - the external flooding fragility analysis, and
    - the plant response logic analysis.
- The analysis identifies and characterizes uncertainties.

The external flooding PRA model reflects the as-built and as-operated plant or the as-designed or the as-to-be built and as-to-be operated plant, as applicable in each stage. Assumptions used in external flood plant response analysis (e.g., screening out external flood initiators and external flood-induced failures, human error event identification and development) should be clearly identified, documented, and tracked to ensure their continued validity across different stages.

C.1.3.14 Other Hazards Probabilistic Risk Assessment Element

This section identifies the hazard group-specific PRA analysis elements, the objectives of those analysis elements, and the characteristics and attributes that are needed for an acceptable other hazards NLWR PRA that addresses all radiological sources, all POSs, and all levels of PRA analysis.

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. An other hazards PRA is performed when other hazards cannot be screened out by a screening analysis. The objective for each other hazards PRA analysis element is briefly described, and the characteristics and attributes needed to achieve the objective are provided below. The other hazards PRA analysis elements are as follows:

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

Screening methods can often be used to show that the contribution of a hazard to risk metrics is not significant. The considerations in this section apply to those hazards identified in Table B-1 of Appendix B that are not screened out based on a screening and conservative analysis for all modeled POSs and sources of radioactive materials. 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 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.1.3.1 through C.1.3.7 of this RG, as necessary.

The objective of the **other hazards analysis** PRA is to establish the frequency of occurrence of different intensities of the hazard being analyzed. The analysis uses a site-specific probabilistic evaluation that is based on current generic or site-specific information. Historical data or a phenomenological model, or a mixture of the two is used in the analysis. The characteristics and attributes needed to achieve the objectives of the other hazards analysis element are as follows:

- The analysis results in the hazard's frequency of occurrence at the site.
- The analysis is based on site-specific data or justified generic data, as applicable.
- The analysis reflects current information.
- The analysis uses historical data, or a phenomenological model, or a mixture of the two.
- The analysis identifies and characterizes the uncertainties related to other hazards.

The objective of the **other hazards fragility analysis** is to perform an evaluation to estimate the fragility or vulnerability of an SSC (or combination of SSCs) whose failure contributes to plant risk. The fragility analysis uses plant-specific information and an accepted engineering method for evaluating failures. The characteristics and attributes needed to achieve the objectives of the other hazards fragility analysis are as follows:

- The analysis is plant-specific.

- The analysis uses SSC-specific information.

- The analysis uses accepted engineering methods.

- Walkdowns or the evaluation of available data and findings of investigations of the plant design and operations information for plants that have not started construction or do not have enough construction complete to perform physical walkdowns focus on all POSs of the plant configuration.

- The analysis identifies and characterizes the uncertainties related to other hazards' fragility.

The objective of the **other hazards plant response analysis** is to develop a model that includes all important initiating events and other important failures caused by the effects of the hazard that can contribute to plant risk. The model is adapted from the internal events PRA model for all modeled POSs and sources of radioactive materials to incorporate unique aspects related to the hazard analyzed that are different from the internal events PRA model. The characteristics and attributes needed to achieve the objectives of the other hazards plant response analysis are as follows:

- The analysis includes all important initiating events related to the hazard analyzed.

- The analysis includes other significant failures that can contribute to plant risk.

- The analysis is adapted from the internal events PRA model for all modeled POSs and radiological sources.

- The analysis incorporates unique aspects related to the hazard analyzed that are different from the internal events PRA model.

- The analysis identifies and characterizes the uncertainties related to other hazards plant response.

C.1.3.15 Event Sequence Quantification Probabilistic Risk Assessment Element

This section identifies the objectives and the characteristics and attributes of the event sequence quantification analysis PRA element for an NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis.

The objective of the event sequence quantification analysis PRA element is to develop a frequency estimate of event sequences and event sequence families at any stage of the plant life cycle, while ensuring that all risk-significant contributors are represented and understood. This element should address all dependencies and demonstrate a complete understanding of PRA uncertainties and assumptions and their impacts on the PRA results. Event sequence quantification integrates the accident progression models and source term evaluation to estimate the frequency of radionuclide releases that can be expected following the accidents. The quantitative evaluation reflects the different magnitudes and timing of radionuclide releases. The characteristics and attributes needed to achieve the objectives of the event sequence quantification analysis are as follows:

- The analysis integrates individual modeling items including the event sequences, system models, event progression phenomena, barrier failure modes, data, HRA elements, dependencies, and recovery actions, and accounts for all functional, physical, and human dependencies.

- The event sequences are quantified using appropriate models and codes and a truncation limit sufficiently low to show convergence of the PRA results.

- The analysis addresses the identification and elimination of circular logic, identification of mutually exclusive event combinations, the use of flag events and modules, and the use of system successes.
- The analysis identifies the risk-significant contributors to the frequency of each risk-significant event sequence and event sequence family.
- Uncertainties in the quantification results are characterized and quantified.

C.1.3.16 Mechanistic Source Term Analysis Probabilistic Risk Assessment Element

This section identifies the objectives and the characteristics and attributes of the mechanistic source term analysis PRA element for an NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis.

The objective of the mechanistic source term analysis is to characterize the radiological release to the environment resulting from each event sequence leading to a release. The characterization includes an identification of risk-significant isotopes to be included in the consequence assessment and data needed to characterize release locations, the physical and chemical form of the released radioisotopes, the time-dependent isotopic release rates to the atmosphere, heat content (or energy) of the carrier fluid, and the data needed to estimate plume buoyancy. The mechanistic source term analysis is sufficient to provide mechanistic source terms for radiological consequence analysis. The computer codes used to perform the analyses for developing the mechanistic source terms are validated and verified for both technical integrity and suitability, and they accurately analyze the phenomena of interest. Qualified personnel who are well trained in the use of the codes perform the calculations. The characteristics and attributes needed to achieve the objectives of a mechanistic source term analysis PRA element are as follows:

- Radionuclide releases are grouped into smaller subsets of representative source terms or release categories.
- Radionuclide releases are assessed for each release category, including consideration of timing, location, amount released, and the radionuclide transport barriers and transport mechanisms.
- Radiological source terms are calculated using appropriate methods or codes.
- Uncertainties in the mechanistic source terms and associated transport phenomena are identified, characterized, and quantified to the extent practical.
- Documentation of the mechanistic source term analysis shall provide traceability of the work.

C.1.3.17 Radiological Consequence Analysis Probabilistic Risk Assessment Element

This section identifies the specific PRA analysis elements, the objectives of those PRA analysis elements, and the characteristics and attributes of the radiological consequence analysis PRA element for an NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis.

The objective for each PRA analysis element is briefly described, and the characteristics and attributes needed to achieve the objective are provided below.[6] These PRA analysis elements are developed using the assumption that exposure to radionuclides released to the atmosphere is the dominant exposure pathway. These elements do not address exposure due to direct radiation from radiological sources within the facility or exposures due to releases of radioactive material to aqueous pathways such as surface water or ground water. The PRA analysis elements for a radiological consequence analysis are the following:

- radionuclide release characterization,
- site characterization,
- meteorological data analysis,
- atmospheric transport and diffusion analysis,
- protective action analysis,
- dosimetry,
- health effects analysis,
- economic factors, and
- conditional consequence quantification.

The objective of the **radionuclide release characterization** is to identify the attributes of the radionuclide release needed to evaluate radiological consequences. It involves the identification of release categories and the development of source term information for each release category. Release category information includes the selection of a representative radiological source term for each release category (as discussed in Regulatory Position C.1.3.16 on mechanistic source term analysis). Source term information, developed as discussed in Regulatory Position C.1.3.16, includes an identification of risk-significant isotopes to be included in the consequence assessment and data needed to characterize release locations, the physical/chemical form of the released radioisotopes, the time--dependent isotopic release rates to the atmosphere, and the data needed to estimate plume rise due to buoyancy, momentum, or both. The characteristics and attributes needed to achieve the objectives of the radionuclide release characterization PRA analysis element are as follows:

- Release categories are defined using acceptable methods (see Regulatory Position C.1.3.16).

- All risk-significant isotopes are included in the radiological consequence analysis.

- Radiological source terms contain information on release locations, the physical/chemical form of the released isotopes, the time-dependent isotopic release rates to the atmosphere, and the data needed to estimate plume rise. If released fractions are used to quantify isotopic release rates, inventories of isotopes are included.

- Radiological source terms used to represent release categories are developed using appropriate methods or codes (see Regulatory Position C.1.3.16).

The objective of the **site characterization** is to provide information on the population distribution and patterns of land use and land cover in the vicinity and region of a site to a distance of 80

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6 Radiological consequence analyses may be performed for a variety of applications, including (but not limited to) assessments used to demonstrate compliance with the dose guidelines of 10 CFR 50.34 (a)(1) and to assist in the preparation of environmental impact statements. The consequence analyses developed under this section are expected to require modification if they are to be used to support those applications. Additional application-specific guidance (e.g., NUREG-0800; NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan" (Ref. 37); and their supporting documents) is available.

kilometers (km) (50 miles (mi)). The location of the exclusion area boundary is identified. The distribution of the population around a site is based on recognized demographic sources, such as census data or local surveys. It is adjusted for population growth and may represent variations in population density surrounding a site. Land use information, such as the distribution of land used for farming and the distribution of water bodies around a site, is based on recognized sources of local or regional geographic information and represents variations in land use and land cover surrounding a site. For PRAs performed prior to selecting a proposed site, the site characterization PRA analysis element is addressed with postulated site data that contain sufficient information to allow comparison of the postulated site data to site data representative of certain points over the life of a proposed facility. The characteristics and attributes needed to achieve the objectives of the site characterization PRA analysis element are as follows:

- The distribution of the population in the vicinity and region of a site to a distance of 80 km (50 mi) is based on recognized demographic sources and adjusted for population growth.

- Land use and land cover information in the vicinity and region of a site to a distance of 80 km (50 mi) is based on recognized sources of local or regional geographic information.

The objective of the **meteorological data analysis** is to evaluate and select the meteorological data used for the atmospheric transport and diffusion analysis. The characteristics and attributes needed to achieve the objectives of the meteorological data analysis PRA analysis element are as follows:

- At least one full annual cycle of hourly meteorological data that are representative of long-term meteorological conditions of the site is compiled. Depending on the application, at least two full annual cycles of hourly meteorological data may be needed. Regulatory Guide 1.23 (Ref. 38) provides information on the amount of meteorological data typically needed at different licensing stages and may be used to justify the amount of meteorological data that is collected and used in the analysis.

- Meteorological data are of acceptable quality and completeness. Regulatory Guide 1.23 provides information regarding assessment of the quality and completeness of meteorological data and may be used to evaluate meteorological data quality and completeness.

- Meteorological data include, at a minimum, data on windspeed, wind direction, atmospheric stability, precipitation, and the depth of the atmospheric mixing layer. Regulatory Guide 1.23 provides information that may be used to determine atmospheric stability from meteorological data.

- Meteorological data sets with missing data are completed by substituting data using interpolation techniques, substitution techniques using data from onsite sources (e.g., from a different tower elevation, nearby onsite locations with similar characteristics), or substitution techniques using data from regional recognized sources (e.g., government weather service stations) where onsite meteorological data are not available.

- For PRAs performed prior to selecting a proposed site, postulated meteorological data are provided that are representative of a reasonable number of sites that have been or may be considered.

- The uncertainties related to the meteorological data analysis are identified and characterized.

The objective of the **atmospheric transport and diffusion analysis** is to perform an evaluation that provides time-dependent air and ground concentrations resulting from a release of radioisotopes. The characteristics and attributes needed to achieve the objectives of the atmospheric transport and diffusion analysis PRA analysis element are as follows:

- An appropriate atmospheric dispersion model is used.
- The analysis uses the meteorological data developed in the meteorological data analysis PRA analysis element.
- The analysis uses a model that includes uniform hourly wind field data from a single representative meteorological tower.
- The analysis includes the selection of dispersion parameters appropriate to the characteristics of the area and distance ranges under consideration. Near-field effects (such as elevated releases of radioactive material, building wake effects such as wake-induced downwash and enhanced diffusion due to near-field wake-induced turbulence, plume meander, and plume rise) are adequately characterized.
- The deposition of airborne material on the ground by wet and dry deposition and the resulting depletion of the airborne material with downwind distance are modeled in a manner that is appropriate for the application.
- The uncertainties related to the atmospheric transport and diffusion analysis are identified and characterized.

The objective of the **protective action analysis** is to characterize the impact of mitigation measures such as evacuation, sheltering, relocation, and interdiction of land, food, or water on doses resulting from releases of radioisotopes. The variability in the responses of offsite populations to releases of radioisotopes may be considered. For PRAs performed prior to selecting a proposed site, protective actions are addressed with postulated data that contain sufficient information to allow comparison of the postulated protective action data to protective actions for a selected site. The characteristics and attributes needed to achieve the objectives of the protective action analysis PRA analysis element are as follows:

- Protective actions that are appropriate for the application are identified and included.
- The analysis is based on recognized sources of protective action guidance such as approved emergency plans and Federal, State, or local guidance. Justification of the sources of protective action guidance used in the analysis is provided when multiple recognized sources recommend different values (e.g., local requirements are more stringent than national requirements, use of international standards in lieu of U.S. standards).
- The analysis of early-phase protective actions includes site-specific consideration of the time at which warning of a release is provided to offsite populations, the delays before the offsite populations either shelter or evacuate, or both, and the speed at which evacuation proceeds. The consideration of these factors is based on recognized site-specific sources such as site-specific emergency plans and site-specific evacuation time estimates.
- Appropriate dose reduction factors associated with occupancy of structures or vehicles are developed and applied.

- The impact of initiating events that may also affect protective actions (e.g., seismic events) is assessed.

- The uncertainties related to the protective action analysis are identified and characterized.

The objective of the **dosimetry** PRA analysis element is to identify the analyses needed to estimate doses to offsite populations arising from airborne and deposited radioisotopes. The characteristics and attributes needed to achieve the objectives of the dosimetry PRA analysis element are as follows:

- Dosimetric quantities (e.g., total effective dose equivalent, equivalent organ doses) to be assessed are identified.

- All relevant short- and long-term exposure pathways (i.e., cloudshine, groundshine, skin deposition, skin absorption, inhalation, ingestion, and resuspension of deposited materials) are identified and included as appropriate for the results of interest.

- The age and gender characteristics of the offsite population are clearly identified.

- The duration of exposure for both acute and chronic exposures is clearly identified.

- Recognized sources of pathway-specific dose coefficients are used to estimate dose from the identified exposure pathways. Dose coefficients are consistent with the dosimetric quantity being assessed (e.g., the use of dose coefficients from Federal Guidance Reports 11 and 12 (Ref. 39 and Ref. 40) are used to for the computation of total effective dose equivalent (TEDE).

- The uncertainties related to the dosimetry PRA analysis element are identified and characterized.

The objective of the **health effects analysis** is to assess the risk of early or latent health effects (either fatal or nonfatal), or both, arising from acute and chronic exposure to released radioisotopes. The characteristics and attributes needed to achieve the objectives of the health effects analysis PRA analysis element are as follows:

- Early and latent health effects are identified and included as appropriate for the application.

- Dose-response models using information from recognized sources are used to estimate the risk of health effects.

- The uncertainties related to the health effects PRA analysis element are identified and characterized.

The objective of the **economic factors** PRA analysis element is to assess the economic impact of releases of radioisotopes, including the economic impact of protective actions taken to limit exposure to released material. For PRAs performed prior to selecting a proposed site, economic factors are addressed with postulated economic data that are representative of a reasonable number of sites that have been or may be considered. This postulated data contains sufficient information to allow comparison of the postulated economic data to economic data for a selected site. The characteristics and attributes needed to achieve the objectives of the economic factors PRA analysis element are as follows:

- Economic factors that are appropriate for the application are identified and included.

- The analysis uses cost parameter values for the time period of interest using regional data applicable to the facility and generic data (as needed) based on recognized sources. Justification is provided for the use of generic data.

- Characterization of economic factors includes consideration of the protective actions taken (e.g., evacuation, temporary or permanent relocation, offsite decontamination, crop disposal, and farmland interdiction).

- Characterization of economic factors includes consideration of the economic characteristics of the region (e.g., the distribution of economic wealth and of economic activities such as farming).

- The uncertainties related to the economic factors PRA analysis element are identified and characterized.

The objective of the **conditional consequence quantification** is to integrate the models and data developed in the preceding technical elements to quantify results of interest. The radiological consequences associated with each release category are quantified. The characteristics and attributes needed to achieve the objectives of the conditional consequence quantification PRA analysis element are as follows:

- Computer codes used for quantification are used within the limits of their applicability.

- Proper code execution is verified.

- Assumptions used to develop the radiological consequence analysis and limitations of the data, models, or computer codes are clearly identified.

- The impact of significant assumptions and limitations on results of interest is adequately characterized.

- Sources of model and parameter uncertainty for each element of the analysis are identified.

- The impact of significant sources of model and parameter uncertainty on results of interest is characterized.

- The impact of variability in meteorological conditions, as reflected in the input parameters related to meteorological observations, on results of interest is quantified.

C.1.3.18 Risk Integration Probabilistic Risk Assessment Element

This section identifies the objectives and the characteristics and attributes of the risk integration PRA element for an NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis. The objectives of the risk integration PRA element are to develop criteria used to determine risk significance, to express overall risk in terms of appropriate risk metrics, and to characterize and quantify the uncertainty associated with the calculated risk metrics.

The objective of determining risk-significance criteria is to identify and justify the criteria by which the risk significance is established for PRA elements such as an event sequence, event sequence families, SSCs, and basic events modeled in the PRA. These risk-significance criteria should be defined consistent with and supportive of the intended application. As part of determining risk-significance criteria, technology-inclusive consequence metrics (e.g., radiological doses, health effects to public) or risk surrogates (e.g., LRF) are used. At a minimum, relative risk-significance criteria should be used to develop the PRA, including but not limited to Fussell-Vesely or Birnbaum importance measures, unless otherwise justified. The characteristics and attributes needed to achieve the objectives of determining risk-significance criteria are as follows:

- The analysis defines consequence metrics (e.g., person-rem, early fatalities, latent health effects, site boundary dose, quantity of radioactive material release) or risk surrogates (e.g., LRF) that allow integration of risks from multiple sources and that support the intended application.
- If the application involves calculation of a PRA baseline risk, the analysis defines and justifies the selection of criteria for establishing the relative risk significance of PRA model elements (e.g., relative risk-significant basic event, relative risk-significant function, relative risk-significant event sequence or event sequence family, relative risk-significant SSC or HFE), accounting for both the frequency and consequences of modeled event sequences.
- If the application can be adequately supported by comparison of risk metrics to fixed targets, the analysis defines and justifies the selection of criteria used to establish the absolute risk significance of PRA model elements, accounting for both the frequency and consequences of modeled event sequences.
- The uncertainties related to the risk-significance criteria are identified and characterized.

The objective of expressing overall risk in terms of appropriate risk metrics is to provide a vehicle by which risk contributions from multiple reactors and other radiological sources can be integrated. The risk metrics used should be consistent with the selected risk-significance criteria and the intended application. The characteristics and attributes needed to achieve the objectives of expressing overall risk in terms of appropriate risk metrics are as follows:

- Information on event sequences and event sequence families is compiled from the event sequence quantification (ESQ) and consequence quantification (RCQ) tasks.
- The integrated risk results are calculated using the risk metric(s) previously defined and event sequences and event sequence families previously compiled.
- Potential differences in level of detail, degree of conservatism, and realism are identified when integrating results for different radiological sources, hazards, or POSs.
- Risk contributions from all sources of radioactive material considered and analyzed in the PRA are included within the scope of the PRA.
- Risk-significant contributors are identified to develop insights from the PRA.
- Methods and codes for risk integration are selected, justified, and applied, accounting for method and code limitations and considering the hazards, POSs, and event sequences that are within the scope of the PRA.
- The uncertainties related to risk metrics are identified and characterized.

The objective of characterizing and quantifying the uncertainties associated with the calculated risk metrics is to provide an understanding of key assumptions and sources of model uncertainties and their potential impact on the results. The characteristics and attributes needed to achieve this objective are as follows:

- A list of key sources of model uncertainties and assumptions for each PRA element in the standard is compiled, and the potential impact of these uncertainties and assumptions on risk results is assessed, including both event sequence family frequencies and consequences. Also, any items that were screened out of the PRA (e.g., hazard groups, POSs, initiating events, event sequences, basic events) are included.

- Uncertainties for event sequence families do not artificially cause these families to be risk significant because of the way event sequences have been grouped into event sequence families.

- A qualitative or quantitative evaluation of the effects of individual sources of uncertainty, or combinations of interest, is performed on each modeled risk metric.

- The uncertainty distribution for the selected risk metric(s) is characterized or calculated.

C.1.3.19 Probabilistic Risk Assessment Documentation

The documentation of the PRA model provides the information necessary to easily reproduce and justify results. 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 by documenting the actual process in the PRA documentation or by reference to existing methodology documents. Sources of both parameter and model uncertainty are identified and documented, and their impact on the results is 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, human reliability model). A key source of uncertainty is one in which the choice of approach or model is known to have an impact on the risk (e.g., total integrated risk, risk of a source, POS, hazard group, frequency of an event sequence or event sequence family, importance measures), or the set of initiating events and event sequences that contribute most to the consequence risk, such that the impact influences a decision supported by the PRA. Assumptions made in performing the analyses are identified and documented along with their justifications to the extent that the context of the assumption is understood. The results (e.g., products and outcomes) from the various analyses are documented. The characteristics and attributes needed for documentation of a given PRA analysis element are as follows:

- 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 and insights.

- The documentation describes the processes used to perform the analyses for each PRA element sufficient to understand the bases of the results of the PRA, including any analysis that is unique to a PRA analysis element.

- The documentation describes the identification and analysis of sources of uncertainty, related assumptions, and reasonable alternatives sufficient to understand the bases of the results of the PRA.

- The documentation describes assumptions and limitations of the PRA due to a lack of data or available plant information.

- The documentation describes the bases for and impact of risk-significant contributors.

- The documentation describes the walkdown process, where applicable, and results of the walkdown. In cases where walkdowns cannot be performed, the documentation describes the evaluation of available data and findings of investigation(s) of the plant design and operations information and the results of that evaluation.

- The documentation describes the results of and bases for each screening analysis, which includes but may not be limited to documenting the selection and application of screening criteria, assumptions used and their validity, the identification and characterization of associated uncertainties, and how the bases for a given screening analysis were confirmed.

C.1.4 Plant Representation and Probabilistic Risk Assessment Configuration Control

Plant representation is defined in terms of how closely the PRA represents the as-designed, as-to-be-built, or as-to-be-operated plant or the as-built and as-operated plant. In general, PRA results used to support an application should be derived from a PRA model that represents the as-designed, as-to-be-built, or as-to-be-operated, plant or the as-built and as-operated plant to the extent needed to support the application. Consequently, the PRA is maintained and upgraded, as needed, to ensure that it represents the as-designed, as-to-be-built, or as-to-be-operated plant or the as-built and as-operated plant, depending on where it is being used in the stages of plant licensing and consistent with the available plant information.

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

Therefore, a process for developing, maintaining, and upgrading an acceptable PRA is established. This process involves identifying and using plant information to develop and modify the PRA, including changes to the plant that necessitate changes to the PRA. The applicant or licensee will consider the cumulative impact of any changes to the plant and PRA model, as needed, on the results of the PRA and on any applications thereof being performed or considered 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 PRA reflects the as-designed, as-to-be-built, or as-to-be-operated plant or the as-built and as-operated plant, as appropriate for the stage of licensing for which the PRA is being used and is as realistic as 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 PRA, the following information is identified, integrated, and used in the PRA:

- plant design information reflecting the normal and emergency configurations of the plant,
- plant operational information about procedures and practices,
- plant test and maintenance procedures and practices, and
- 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 be gained only by direct observations. At the design and initial licensing stages, plant walkdowns are not possible; however, in these cases, available data on the plant design and operations information should be evaluated.

The sources of information that should be used in the development of a PRA include, but are not limited to, those that provide the following types of information:

- the safety functions relied on to maintain the plant in a safe, stable state and prevent damage to radionuclide transport barriers and releases of radioactivity;
- identification of those SSCs that are credited in the 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;
- SSCs' capabilities (flows, pressures, actuation timing, environmental operating limits);
- spatial layout, sizing, and accessibility information related to SSCs relied on for prevention and mitigation of releases of radioactive material;
- other design information needed to support the modeling of the plant in the PRA for any stage of the licensing process;
- the design margins addressed by the capabilities of the SSCs;
- operating environment limits of equipment;
- expected thermal-hydraulic plant response to different operational states of equipment (such as for establishing success criteria); and
- other relevant engineering information (e.g., relevant information for a related technology, generic industry information) needed to support the modeling of the plant in the PRA for any stage of the licensing process.

For plants that have operational experience, the sources of information used in the development of a PRA should also include, but are not limited to, those that provide the following types of information:

- historical information related to the maintenance practices and experience at the plant; and
- information on planned and typical unplanned tests and maintenance activities and their relationship to the status, timing, and duration of equipment availability.

The information sources listed above should be accurate and representative of the design and operating characteristics and have an *adequate* technical basis to support the associated analysis.

As plant design, construction, and operations progress over time, its associated risk may change for the following reasons:

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

Therefore, to ensure that the PRA represents the risk of the as-designed, as-to-be-built, or as-to-be-operated plant or the current as-built and as-operated plant, depending on the stage of the licensing process, the PRA is maintained and upgraded over time. COL holders should meet all applicable requirements in the ASME/ANS NLWR PRA standard, 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). The characteristics and attributes of an acceptable process for maintaining and upgrading a PRA are as follows:

- The process is capable of monitoring PRA inputs and the collection of new information affecting the PRA.

- The cumulative impact of pending plant changes is considered.

- The process includes maintaining the configuration control of computer codes used to support the PRA.

- The process establishes when the PRA model should be updated based on new information or new models, techniques, and tools.

- A peer review is performed after the PRA is upgraded.

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

One acceptable approach to demonstrate conformance with the regulatory positions in Section C of this RG is to use a national consensus PRA standard or standards, as endorsed by the NRC staff with exceptions, that address the scope of the PRA. ASME and ANS have issued the ASME/ANS NLWR PRA standard. This standard provides process and technical requirements for a PRA of an NLWR that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis (i.e., from initiating event to radiological consequences). National consensus PRA standards establish requirements for what an acceptable PRA should include to satisfy the applicable regulations in 10 CFR Part 50 and 10 CFR Part 52. However, these PRA standards do not address how to meet the requirements for an acceptable PRA. Because the joint ASME and ANS national consensus PRA standards use the term "requirement," "require," and other similar mandatory language, the staff's endorsement, including its exceptions, mirrors this language. The use of this language in this RG does not mean that compliance with this RG is mandatory or is 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 Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality and Information Requests" (Ref. 41).

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 for this purpose, a peer review is important for determining whether the underlying purpose of requirements in the national consensus PRA standard are met, as endorsed by the NRC with exceptions, so that it can be demonstrated that the PRA model conforms to Regulatory Position C.1. Regulatory Position C.2.2 presents the staff position on the use of PRA peer reviews to this effect, including staff endorsement with exceptions, 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, this represents an acceptable peer review process. The NRC staff considers the PRA acceptable for supporting the application based on the results of the peer review, resolution of the Facts and Observations (F&Os), and how the PRA conforms to the requirements in the national consensus PRA standards. Use of the ASME/ANS NLWR PRA standard and NEI 20-09, Revision 1, as endorsed by the NRC in this RG, reduces the need for an in-depth staff review of the PRA.

C.2.1 National Consensus Probabilistic Risk Assessment Standards

National consensus PRA standards provide requirements for an acceptable PRA. However, it is recognized that a PRA may not always need to satisfy each technical requirement to the same degree. The NLWR PRA standard features two capability categories (CCs), CC-I and CC-II, which are used to distinguish between greater and lesser scopes, levels of detail, plant representation, and realism needed for a given technical requirement. When a supporting requirement (SR) provides a different requirement for each capability category, the CC-I requirement generally fosters identification of the most risk-significant event sequences at a functional or systemic level. The CC-II requirement fosters the development of a realistic assessment of risk. The CC achieved for the different technical requirements may vary. In terms of the staff position in this RG, this variation can range from the minimum capability needed to meet the characteristics and attributes for each PRA element (i.e., CC-I) to the minimum capability needed to meet current good practice (i.e., state-of-practice) for each PRA element (i.e., CC-II). Further, the capability category that needs to be met for each technical requirement depends on the application. In general, the staff anticipates that meeting CC-II should result in an acceptable scope, level of detail, and realism for most applications. However, for some applications, CC-I may be acceptable for some requirements.

The requirements in an ASME/ANS PRA standard are either process related or technical. Process-related requirements address the process for development, application, maintenance and upgrade, and peer review of a PRA and its results (including resolution of the F&Os) used in support of an application. The technical requirements address the elements of the PRA and what is necessary to acceptably perform that element.

For process-related requirements, the purpose 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 consensus standard are applied more than once in developing the PRA model. For example, the requirements for systems analysis in an internal events, at-power PRA apply to all systems modeled, and certain data requirements apply to all parameters for which estimates are provided. If the requirement has been met for the majority of the systems or parameter estimates, and any mistakes or oversights are identified as isolated instances, the staff may consider the requirement 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, instances of noncompliance with the requirements in an ASME/ANS PRA standard are to be (1) rectified or demonstrated not to be relevant to the application and (2) documented accordingly.

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 PRA element and are intended to achieve the objective of the PRA element. HLRs are defined in general terms, should be met regardless of the CC, and accommodate different approaches. SRs are defined for each HLR and are the minimum 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 related process requirements. If any SRs are determined to be inapplicable, justification for such a conclusion should be documented and peer reviewed. All SRs related to new developed methods (NDMs) should be evaluated during peer reviews of NDMs.

If different requirements are used, other than those in an established national consensus PRA standard, then it should be demonstrated how these different requirements are reasonable and acceptable for assessing and establishing what an acceptable PRA should include, as well as what acceptable processes should include. It should also be demonstrated how the different requirements meet the regulatory positions in Section C of this RG.

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, have been met. An acceptable peer review approach is one that follows an established process and is done by qualified personnel, documents the results, and identifies both strengths and weaknesses of the PRA. The ASME/ANS NLWR PRA standard requires a peer review to be performed on the PRA model, any PRA upgrades, and the use of any NDMs. A peer review methodology (i.e., process) is documented in the industry-developed peer review guidance documents.

This section of the RG endorses on a trial basis the process for performing PRA peer reviews provided in NEI 20-09, Revision 1, as one acceptable approach for determining whether a PRA meets the requirements in the ASME/ANS NLWR PRA standard. In addition to the guidance in NEI 20-09, Revision 1, the ASME/ANS NLWR PRA standard also presents the general requirements for a peer review to determine whether the PRA methodology and its implementation meet the technical requirements in the standard. The NLWR PRA standard, as endorsed by the NRC with exceptions, includes requirements for establishment of a peer review process, PRA peer review team qualifications, and documentation.

The NRC staff reviewed NEI 20-09, Revision 1, to determine whether the peer review process described therein is acceptable for establishing the acceptability of a PRA. For the reasons given below, the staff finds that the guidance in NEI 20-09, Revision 1, is acceptable in that regard and endorses NEI 20-09, Revision 1, without exception. The ASME/ANS NLWR PRA standard contains requirements for the performance of an acceptable peer review process. The staff reviewed the requirements and takes no exceptions to them. The process described in NEI 20-09, Revision 1, is considered to be acceptable for a peer review performed for a PRA representing any stage of a plant's life cycle, recognizing the varying level of detail in the PRA to account for effects such as the certainty of the design and the availability of plant-specific data. The peer review process endorsed in this section can accommodate any scope of PRA peer review, as defined by the user, including a focused-scope peer review for a PRA upgrade. The following are important aspects of NEI 20-09 the staff considered when evaluating the guidance document for endorsement.

As part of an application, an applicant or licensee should describe the measures it has taken to ensure that the design-specific or plant-specific PRA is acceptable for its intended use. The measures may include items such as any self-assessments and peer reviews against the ASME/ANS NLWR PRA standard, as well as any actions taken to address self-assessment and peer review results.

When performed prior to the application, the peer review provides findings and observations on PRA completeness and acceptability, including consideration of the scope, level of detail, conformance to a consensus PRA standard, plant representation of the PRA model, the assumptions made in the development of the results, and the uncertainties that impact the analysis. If a peer review has not been performed and the applicant's justification fails to give the staff adequate confidence in the PRA models, results, and insights, then an in-depth staff review is warranted. An in-depth staff review will assess the applicant's PRA against the PRA elements and the staff positions described in this RG to determine the PRA's acceptability. Because key assumptions, logic modeling, and modeling parameters can significantly impact the PRA results and insights, staff review of their acceptability is necessary to ensure that the PRA yields reasonable and acceptable information that can be relied on when making risk-informed regulatory decisions.

In general, an acceptable peer review process should identify the necessary steps to compare the PRA against established requirements and criteria (e.g., technical requirements defined in the NLWR PRA standard). As part of this process, the PRA models are compared against the plant design and procedures, if available, to validate that the models reflect the as-designed, as-to-be-built, as-to-be-operated plant, or the as-built and as-operated plant. Additionally, the peer reviewers perform independent walkdowns, if possible, to confirm PRA inputs, especially for external hazard PRAs. Assumptions are also reviewed to determine whether they are appropriate and to assess their impact on the PRA results and insights. The PRA results are checked for fidelity with the model structure and for consistency with the results from PRAs for similar plants, if available. Finally, the peer review process also examines the procedures or guidelines established for upgrading and updating the PRA to reflect changes in plant design, operation, or experience.

The peer review team qualifications are important for determining the credibility, independence, and acceptability of the team members. To avoid any perception of a technical conflict of interest, the members of the peer review team should be prohibited from peer reviewing any portion of the PRA on which they have performed or supervised efforts. Each member of the team should have technical expertise in the PRA elements they review, including experience in the specific methods used to develop elements of the PRA. 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 PRA. The staff recognizes that when an applicant conducts a peer review or seeks an independent assessment of the acceptability of PRAs performed during the preoperational stage, the independent review team will likely not have specific and detailed knowledge of all aspects of the design, but members should be familiar with the general design and operating philosophy based on the design and operating guidance available for that stage of the plant life cycle.

Chapter 4 of NEI 20-09, Revision 1, and Section 6.2 of the ASME/ANS NLWR PRA standard provide specific qualifications that peer review team should meet. The staff acknowledges that a requirement of absolute independence coupled with the need for adequate technical expertise can be difficult to achieve in some situations. However, the staff emphasizes that a peer review team should have the following attributes, which are listed in Section 4.4 of NEI 20-09:

- independent of the PRA being reviewed (e.g., team members should be prohibited from peer reviewing any portion of the PRA for which they have performed or supervised work),

- experienced in the stage of plant life cycle (i.e., phases as referred to in NEI 20-09 of the PRA being reviewed),

- knowledgeable about the specific reactor technology and its design used for the development of the PRA under review, and

- familiar with relevant regulatory guidance for the regulatory activity under consideration.

The staff agrees with the discussion in NEI 20-09, Revision 1, that because of the unique design and safety features of NLWRs, host user personnel with detailed knowledge of reactor design and analysis should support the peer review process.

Peer review documentation is essential for providing the necessary information to ensure that the peer review and the results of the peer review are traceable, and the bases of the results of the peer review 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 PRA element and the PRA update process should be described. This should include an assessment of the importance of any identified deficiencies in the PRA and its results and how these deficiencies should be addressed and resolved.

A peer review of a PRA evaluates the PRA models for all radiological sources, all hazards, all POSs, and all levels of analysis needed for a given application. The peer review also examines the associated configuration control process, including processes for maintaining and upgrading the PRA. As part of quality assurance reviews of PRA documentation, the peer review should consider the principal elements of the types of quality assurance reviews performed in accordance with 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." This includes consideration of the following:

- the use of qualified reviewers,
- the use of reviewers who are independent of the original PRA development and any relevant upgrades of the PRA,
- the list of issues to be addressed in the PRA, and
- documentation of the review conclusions.

Chapter 5 of NEI 20-09, Revision 1, identifies four main steps in the overall peer review process:

- preparatory activities,
- offsite review,
- onsite consensus review, and
- documentation of peer review results, including interaction with host user.

Figure 1-1, "PRA Peer Review Process Flow Chart," in NEI 20-09, Revision 1, depicts a PRA peer review framework outlining the approach and process steps used in a peer review for an individual PRA. The staff finds this process, in total, to be acceptable for a given peer review.

The PRA peer reviewers assign CCs to each of the SRs of the various elements of the PRA standard to judge whether the PRA meets the SRs in the NLWR PRA standard within the scope of the review. A summary of the SR review is then provided for each HLR. The SRs define the minimum requirements necessary to meet each CC. Some of the SR action statements apply to only one CC, while others cover both CCs. When an action statement spans both categories, it applies equally to each CC. A PRA is considered to have met an HLR if the PRA meets all the applicable SRs under that HLR. The peer review team should determine that an SR is not met when a preponderance of evidence demonstrates that the minimum requirements in an SR at a particular CC are not met.

The peer review should identify any issues that impact the acceptability of the PRA and document these problems in an F&O. The F&Os specify the PRA element and SR of concern and describe the level of compliance with that SR in the PRA. The level of significance of each F&O should be characterized as one of the following:

- finding—an issue or discrepancy that is necessary to address to ensure the technical adequacy of the PRA, the capability of the PRA, or the robustness of the PRA update process;

- suggestion—an observation considered desirable to maintain maximum flexibility for applications and consistency with industry practices;

- best practice—an observation of practices that utilities throughout the industry would want to emulate; and

- unreviewed analysis method—an observation regarding the use of methods that are new or beyond the expected expertise of the review team.

A finding F&O should be written for an SR assessed at CC-I when the SR is being assessed against CC-II.

The product of a peer review is a written report documenting the details, findings, observations, conditions, and results of the review. The peer review team should document the results of the review following the guidance in Section 9 of NEI 20-09, Revision 1.

A follow-on peer review is performed after the initial peer review of the PRA has already been conducted and at least the F&Os classified as “findings” from the previous peer review have been identified and addressed. A follow-on peer review should be conducted after a PRA has been upgraded. A follow-on peer review’s scope can be as narrow as a single individual SR within a PRA element, or as expansive as a peer review of the entire PRA for a given hazard.

When an NDM is used in a PRA, the method should be peer reviewed to determine its acceptability. The peer review of an NDM assesses whether the method meets a particular set of technical requirements and, consequently, can be used to support the PRA. After an NDM has been successfully peer reviewed, its implementation in a PRA is considered to be an upgrade and should therefore be subject to an implementation peer review. An acceptable approach to performing a peer review for an NDM is described in NEI 20-09, Revision 1. In particular, NEI 20-09, Revision 1, 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 remains open, a licensee or applicant may not use the method in a PRA supporting risk-informed licensing applications. Because the peer review and F&O finding-level are adequate to determine the acceptability of an NDM, the NRC staff has determined that this provision establishes adequate control over the use of an NDM. If open F&Os from an NDM peer review cannot be successfully closed through an NRC-endorsed closure process, the NDM could be submitted to the NRC to determine its acceptability.

The staff recognizes that for certain types of NDMs (e.g., fundamentally novel methods or technologies), some direct review by the NRC staff may be warranted. Submitted applications that use NDMs with open F&Os are subject to review by the NRC to determine the acceptability of the method, 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 acceptability of an NDM are listed below:

- The purpose and scope of the NDM are clearly stated.

- The NDM is based on sound engineering and science relevant to its purpose and scope.

- The data (note that data can be numeric or nonnumeric) are relevant to the NDM, technically sound, and properly analyzed and applied.

- Uncertainties in the NDM are characterized. Sources of model uncertainties and related assumptions are identified.

- The results of the NDM are reproducible, reasonable, and consistent with the assumptions and data, given the purpose and scope of the method.

- The documentation of the NDM provides traceability of the work and facilitates incorporation of the method in a PRA model.

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

This section of the RG provides guidance to applicants and licensees on an approach acceptable to the NRC staff to demonstrate the acceptability of a PRA and its results used to support an application. For all applications, the PRA-related information provided in the submittal should do the following:

- Describe the PRA's scope–such as, but not limited to, consideration of all radiological sources, hazards, plant operating states and levels of analysis– level of detail, and degree of plant representation.

- Demonstrate that the PRA has been developed and used in a technically acceptable manner, including the appropriateness of the assumptions and approximations used in developing the PRA.

- Identify the application-specific acceptance criteria and demonstrate that they have been met.

The following sections provide more detailed guidance on each of these aspects of the staff assessment. PRA acceptability for a given risk-informed activity is determined in the context of the staff positions in this RG and relevant application-specific regulatory guidance.

C.3.1 Probabilistic Risk Assessment Scope, Level of Detail, and Degree of Plant Representation

The scope of a PRA needed to support an application will depend on the application-specific regulatory requirements, and the acceptability of the scope will be measured in terms of whether the applicant or holder of a license, certification, or permit meets those requirements. Application-specific guidance documents are expected to provide direction on meeting such requirements.

For plants in the preoperational stages of the plant life cycle, the PRA and its results used to support an application are expected to reflect the as-designed, as-to-be-built, or as-to-be-operated plant. For operating plants, the PRA should reflect the as-built and as-operated plant. When used for risk-informed decision-making, the PRA should always reflect the best available information for the plant. For most applications, an applicant or holder of a license, certification, or permit should address all radiological sources, all hazards, all POSs, and all levels of analysis, as discussed in Regulatory

Position C.1.1 of this RG. The staff will assess the appropriateness of the justification for any deviations from this scope.

C.3.2 Development and Use of an Acceptable Probabilistic Risk Assessment

The staff positions in Regulatory Positions C.1 through C.1.4 represent the minimum capability the staff has determined that a PRA should possess to support risk-informed regulatory activities. When this RG is used to determine the acceptability of a PRA, all staff positions in this RG should be met for a more efficient review by the staff and for a PRA to be considered acceptable. One acceptable approach for demonstrating conformance with regulatory positions in this RG and thereby reducing the need for an in-depth staff review of the PRA is to use an NRC-endorsed national consensus standard during the development of the PRA and to have the PRA peer reviewed through an NRC-endorsed process. The ASME/ANS NLWR PRA standard provides the technical requirements for this purpose. If the ASME/ANS NLWR PRA standard is used, as endorsed by the NRC in Appendix A to this RG, Regulatory Positions C.1 through C.2 are considered to be met. Deviations from a staff endorsement of a PRA technical requirement or a staff position are evaluated for acceptability on a case-by-case basis.

When the exceptions raised by the staff are taken into account, the national consensus standard or PRA peer review process in question is considered to be acceptable for its intended purpose. If the PRA is demonstrated to have met the requirements of these documents, with attention paid to the NRC's exceptions, it can be assumed that the analysis is technically correct. Thus, the staff should be able to focus more on the assumptions and approximations associated with the application. In that way, the need for a detailed review by NRC staff of the PRA should be reduced. 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 is such that no changes occur in the risk contributors.

As discussed in Regulatory Position C.2.2 of this RG, a peer review is performed to determine whether the requirements established in a national consensus standard, as endorsed by the NRC with exceptions, have been met. The peer review includes assessing the appropriateness of assumptions and approximations used in the PRA. This helps ensure that the technical aspects of the PRA have been developed in a technically correct manner and consistent with industry practices. In addition to assessing the PRA against an NRC-endorsed national consensus standard, the peer review also assesses whether the methods used to develop the PRA were applied correctly and that the probabilities and frequencies used are estimated consistently with the definitions of the corresponding events in the PRA logic model and based on the best information available. The PRA model is compared against the plant design and procedures to validate that it reflects the as-designed, as-to-be-built, or as-to-be-operated plant or the as-built and as-operated plant, depending on the stage of the plant life cycle. The results of a peer review should be used to help ensure that the PRA was developed in a technically correct manner as it relates to whether the technical requirements in a national consensus standard have been met.

PRA models rely on the use of approximations and assumptions that may reflect a lack of information, that may be used to address uncertainties related to specific modeling issues, or that make the models more tractable. The impact of these assumptions and approximations on the results used in support of the application should be understood. For a given PRA, different analysts may use different assumptions and approximations but still be consistent with the requirements of the national consensus standard, or the assumptions and approximations may be acceptable under the guidelines of the peer review process, as endorsed by the NRC. The choice of a specific assumption or a particular approximation or assumption is considered to be key if it can influence the results of the PRA and, therefore, influence the application under consideration. For each application that uses this RG to meet regulatory requirements, the assumptions and approximations relevant to that application and those that are key to that application are identified. The key assumptions are used to identify sensitivity studies that inform the decision-making associated with the application. When a key assumption is shown to be consistent with a consensus method or approach, that key assumption is not likely to be subject to additional sensitivity studies in the context of an application, as determined on a case-by-case basis. Based on an understanding of how the PRA model is to be used to achieve the desired results, the licensee should have identified the parts 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 event sequences. For some applications, this may be some subset of all items in the PRA, but for others (e.g., risk-informing the scope of special treatment requirements), all parts of the PRA model may be relevant.

The current state-of-practice in PRA technology reflects that there are issues where there is no consensus on the method of analysis. However, in the context of risk-informed regulatory decisions, a method or model approach that the NRC has used or accepted for the application is considered to be a consensus method or consensus model. A consensus method or model may have a publicly available, published basis and may have been peer reviewed and widely adopted by an appropriate stakeholder group.

Assurance that the PRA and its results used to support an application have been developed and used in a technically correct manner indicates that (1) the PRA model supporting the application represents the as-designed, as-to-be-built, or as-to-be-operated plant or the as-built and as-operated plant. This assurance 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 and it correctly reflects the dependencies among 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 and based on the best information available.

The applicant or holder of a license, certification, or permit should demonstrate that the PRA model represents the as-designed, as-to-be-built, and as-to-be-operated plant or the as-built and as-operated plant, as dictated by the application. Demonstrating this can be achieved through (1) the establishment of a PRA configuration control process that includes provisions for updating the model periodically to reflect changes that impact the significant event sequences, and (2) using a national consensus standard, as endorsed by the NRC. Additionally, PRA self-assessments and peer reviews that follow an approved process should be used, as endorsed by the NRC, to demonstrate how the PRA meets the NRC-endorsed requirements in a national consensus standard. As discussed in Regulatory Position C.2.2 and its subsections, NEI 20-09, Revision 1, provides current industry guidance on self-assessments and peer review, which is endorsed in this RG.

C.3.3 Application-Specific Acceptance Criteria and Guidelines

Application-specific guidance documents identify the applicability of acceptance criteria or guidelines for a given application. Such guidance documents should address the PRA results needed to compare against the acceptance criteria or guidelines and how the comparison should be done. By following this guidance, an applicant or holder of a license, certification, or permit should be able to readily demonstrate the applicability of the application-specific acceptance criteria or guidelines inherent to the application.

More broadly, the Commission articulated in its policy statement "Safety Goals for the Operation of Nuclear Power Plants" (51 FR 28044; August 4, 1986 as corrected and republished at 51 FR 30028; August 21, 1986) two qualitative safety goals, which are supported by two quantitative goals (i.e., the QHOs). These are discussed in Regulatory Position C.1.1 of this RG. The Commission's Safety Goals policy statement expresses its views on the level of risks to the public health and safety that the nuclear industry should strive to meet for nuclear power plants. If the safety goals and QHOs are not already used as acceptance criteria or guidelines for a given application, the applicant or holder of a license, certification, or permit should demonstrate how the application meets them.

C.4 Probabilistic Risk Assessment Documentation in Support of a Regulatory Decision

PRA documentation should be sufficient to allow the staff to determine the acceptability of the PRA and the PRA results used to support the application under consideration. Thus, the PRA documentation should include information necessary for the staff to gain a full understanding of the technical bases of the PRA and how the assessment and its results are used to support the application.

While developing an application, the applicant or holder of a license, certificate, or permit documents the PRA model and the analyses performed to support the application under consideration. This PRA documentation comprises both archival (i.e., available for audit or inspection) and submittal (i.e., submitted as part of the risk-informed request) documentation. Archival PRA documentation may be required on an as -needed basis to facilitate the NRC staff's review of the application.

In general, all PRA documentation should be retrievable, complete, and updated as needed based on an approved configuration control process to help ensure that the PRA and its results used to support a given application represent the as-designed, as-to-be-built, and as-to-be-operated plant or the as -built and as -operated plant. Application-specific guidance indicates how to meet documentation submittal requirements for a specific application. If available, application-specific guidance may provide specific guidance on archival PRA documentation and information that should be included with a submittal. However, in the absence of such application-specific guidance, Regulatory Positions C.4.1 and C.4.2 provide generic characteristics and attributes of archival and submittal documentation that should otherwise be achieved.

C.4.1 Archival Probabilistic Risk Assessment Documentation

Certain generic characteristics and attributes of archival PRA documentation are fundamental to the staff's assessment of PRA acceptability and should be achieved during the creation of that documentation by an applicant or holder of a license, certification, or permit in the absence of application-specific guidance. As part of achieving these characteristics and attributes, the archival PRA documentation should include a detailed description of the following:

- The process used to determine the acceptability of the PRA, including a description of how the staff position in this RG is met, should be included. The description should state whether a national consensus standard was used and whether the technical requirements in that standard are met.
    - o If a national consensus standard was used as part of demonstrating PRA acceptability, the documentation should show that the PRA was developed consistently with that standard, as endorsed by the NRC in this RG.
    - o If a national consensus standard was not used or was used in part to demonstrate PRA acceptability, justification should be developed for each requirement from the related national consensus standard that is not met to explain why not meeting the requirement is acceptable. This justification should include sensitivity studies demonstrating that the event sequences or significant contributors to the application are not adversely impacted.

The documentation should also describe the following:

- the methodology used to assess the risk of the application, including details of how the risk was quantified and identification and justification of all assumptions and approximations used to develop or evaluate the PRA;

- SSCs, operator actions, and plant operational characteristics affected by the application, including the cause-effect relationships among SSC behavior, operator actions, and plant operational characteristics;

- how the cause-effect relationships are mapped onto the PRA elements;

- the PRA results that will be used to compare against the applicable acceptance criteria or guidelines including how the comparison was performed;

- the scope of risk contributors (hazard groups and modes of operation) included in the PRA to support the application;

- the results of the peer reviews of the PRA, PRA upgrades, and use of NDMs, and the results of F&O independent assessments (as discussed in Regulatory Position C.2.2), to include the resolution of all of the peer reviews (i.e., PRA, PRA upgrades, and use of NDMs) and F&O independent assessments; the results should be documented such that it is clear why each requirement is considered to have been met;

- the processes for maintaining and upgrading the PRA and the use of NDMs, including the cumulative history of those activities such as the results of peer reviews that were performed as a result of a PRA upgrade or the use of an NDM;

- the resolution of the peer review findings for the NDMs if the PRA under consideration includes NDMs that have open finding-level F&Os from the technical assessment peer review against the NDM criteria, as endorsed in Appendix A to this RG. This should also include information 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 Appendix A to this RG. Such information should provide, 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; and

- the implementation of an 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 PRA under consideration.

C.4.2 Submittal of Probabilistic Risk Assessment Documentation

In addition to the characteristics and attributes of archival PRA documentation described in Regulatory Position C.4.1, submittal PRA documentation likewise has generic characteristics and attributes that are fundamental to the staff's assessment of PRA acceptability for a given application. In the absence of application-specific guidance on submittal PRA documentation, these generic characteristics and attributes should otherwise be achieved during the creation of that documentation by an applicant or holder of a license, certification, or permit. As part of achieving these generic characteristics and attributes, the submittal PRA documentation should include the following:

- demonstration that the PRA model represents the as-designed, as-to-be-built, and as-to-be-operated plant or the as-built and as-operated plant, which includes:
    - identification of permanent plant changes, such as design or operational practices, that have a potential impact on the PRA but that have yet to be represented in the PRA;
    - justification for why a permanent plant change does not impact the PRA results and insights used to support the application. This justification could be in the form of a sensitivity study demonstrating that the event sequences or contributors significant to the application were not impacted;
    - justification that the PRA has been performed in such a way that its results are acceptable, commensurate with the level of maturity of the design. This should include any commitments for updating the PRA to ensure that it reflects the plant design for and is consistent with the PRA's intended use in the application.
- description of the appropriateness of key assumptions, approximations, and sensitivity studies thereof relevant to the results used in the application.
- justification of the appropriateness of a given portion of the PRA that does not meet an SR or meets a CC lower than that deemed appropriate for the application as specified in the application-specific guidance.
- description of the appropriateness of PRA model upgrades, including the use of NDMs, for the application under consideration.

D. IMPLEMENTATION

The NRC staff may use this trial use regulatory guide as a reference in its regulatory processes, such as licensing, inspection, or enforcement. The purpose of a trial use regulatory guide, such as this one, is to allow early use prior to general implementation. As a result, the staff anticipates continuing to evaluate the positions in this regulatory guide. Therefore, staff may withdraw or add positions from this trial use guide after the trial use period ends if a position is determined to be not acceptable. Moreover, this trial use regulatory guide does not establish a staff position for purposes of backfitting as that term is defined in 10 CFR 50.109, “Backfitting,” and as described in NRC Management Directive (MD) 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests”, 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.” Any changes to this trial regulatory guide prior to staff’s adoption of a final regulatory position will not be considered to be backfits as defined in 10 CFR 50.109. This trial use regulatory guide also does not constitute forward fitting as that term is described in MD 8.4.

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

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8 Copies of American Society of Mechanical Engineers (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/. Copies of American Nuclear Society (ANS) standards may be purchased from the ANS Web site (http://www.new.ans.org/store/); or by writing to: American Nuclear Society, 555 North Kensington Avenue, La Grange Park, Illinois 60526, U.S.A., Telephone 800-323-3044.

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APPENDIX A

NRC REGULATORY POSITION ON ASME/ANS RA-S-1.4-2021

Introduction

The American Society of Mechanical Engineers (ASME) and the American Nuclear Society (ANS) have published ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants" (Ref. 1). The standard gives the requirements[1] for use of probabilistic risk assessments (PRAs) to support risk-informed decisions for non-light water reactor (NLWR) nuclear power plants (NPPs) and prescribes a method applying these requirements for specific applications. The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed ASME/ANS RA-S-1.4-2021 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-S-1.4-2021 is categorized as "no objection," "no objection with clarification," or "no objection subject to the following qualification." These categories are 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-S-1.4-2021 includes the following risk assessment technical requirements:

- Plant Operating State Analysis
- Initiating Event Analysis
- Event Sequence Analysis
- Success Criteria Development
- Systems Analysis
- Human Reliability Analysis
- Data Analysis
- Internal Flood PRA
- Internal Fire PRA
- Seismic PRA
- Hazards Screening Analysis
- High Wind PRA
- External Flooding
- Other Hazards PRA
- Event Sequence Quantification

[1] Because the PRA consensus standards use the terms "requirement," "require," and other similar mandatory language, the staff's endorsement, including staff exceptions, mirrors this language. However, the use of this language in this RG does not imply that this RG imposes any regulatory requirement or suggest that these standards are the only way to meet the statutory and regulatory requirements.

- Mechanistic Source Term Analysis
- Radiological Consequence Analysis
- Risk Integration

The NRC understands that the nonmandatory appendices (NMAs) provided in ASME/ANS RA-S-1.4-2021 are not requirements. Rather, based on discussions with the JCNRM, the NRC understands that the JCNRM's underlying purpose for providing the NMA notes and commentary was to help ensure that PRA analysts are apprised of certain known characteristics, challenges, and issues associated with the NLWR PRA model. While some of the discussion includes "primer-like" information, the language should not be viewed as prescriptive. The analyst should not interpret NMAs as limiting flexibility in the conduct of the technical analyses, or in the application of expert and engineering judgement. A broad range of tools, techniques, implicit/explicit analysis, and judgement may be required to address the diverse modeling required. With respect to the NMAs that provide notes on specific supporting requirements (SRs), the NRC understands that the JCNRM's underlying purpose was to clarify the intent of a supporting requirement (SR), explain jargon that might be used in an SR, and/or provide examples of analysis approaches that would meet the intent of the SR.

Accordingly, this RG does not endorse or approve for use any of the NMAs contained in ASME/ANS RA-S-1.4-2021. This lack of endorsement or approval for use does not necessarily mean that the NRC disapproves the substance nor limits the use of the information provided in the NMAs or that the NRC is limiting the use of that information. Applicants and licensees should meet the applicable requirements of ASME/ANS RA-S-1.4-2021 regardless of whether they use some or all of the information provided in the NMAs.

Tables A-1 through A-22 provide the staff's position on each of the requirements of the standard. The tables present the staff's concern (issue) and the staff's proposed resolution. In the proposed staff resolution, the staff clarification or qualification of the requirement is indicated in either bolded text (i.e., **bold**) or strikeout text (i.e., ~~strikeout~~); that is, the tables provide the additions or deletions to the requirement (as written in the ASME/ANS RA-S-1.4-2021 PRA standard) necessary for the staff to have no objection. Italic text (i.e., *italic*) is used to denote an NRC commentary that does not involve any changes to the requirement.

As stated in Management Directive (MD) 6.6, "Regulatory Guides" (Ref. 2), in a regulatory guide, endorsement of a document, or part of a document, means that the staff has evaluated the material and has found that it is acceptable for use, either in whole or in part, and allowed for use by licensees as discussed within the RG. The staff endorsement on this RG does not intend to provide a new interpretation of requirements in the ASME/ANS NLWR PRA standard, as necessary interpretations are provided by the standards developing organization, nor to change the text of the standard. The staff position provides resolutions to identified issues on using an acceptable PRA to support a regulatory application.

**Table A-1. Staff Position on ASME/ANS RA-S-1.4-2021,
NLWR Standard Introduction, Acronyms and Definitions, and Risk Assessment
Application Process**

Index No.	Issue	Position	Resolution
Document-wide			
General	The phrase “advanced non-LWR” is used throughout the standard but is not defined.	Clarification	*The standard may be applied to any non-light water reactor (NLWR), regardless of whether the NLWR incorporates one or more of the attributes listed in the Commission’s “Policy Statement on the Regulation of Advanced Reactors” (73 FR 60612; October 14, 2008).*
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: No staff position is provided on this reference. The staff neither approves nor disapproves of information contained in the referenced document.*
Section 1: Introduction			
Section 1.1 through Section 1.12	-----------------------	No objection	-------------------------------
Section 2: Acronyms and Definitions			
Section 2.1			
Acronyms	-----------------------	No objection	-------------------------------
Section 2.2			
Feasibility	In the context of operator actions, several high-level requirements and supporting requirements refer to “feasibility”; however, this term is not defined in the NLWR PRA standard. Feasibility	Qualification	*Add the following definitions*: (1) **Feasibility assessment—the qualitative consideration of whether the operator action can be accomplished, considering several performance-shaping factors.** (2) **Feasible—an operator action that can be credited in**

Table A-1. Staff Position on ASME/ANS RA-S-1.4-2021, NLWR Standard Introduction, Acronyms and Definitions, and Risk Assessment Application Process

Index No.	Issue	Position	Resolution
	is a continuous step in the human reliability analysis (HRA) process. Definitions related to feasibility are needed to eliminate ambiguous interpretation.		**a PRA model if the action has met all the feasibility assessment criteria (see supporting requirement HR-H2).**
Section 3: Risk Assessment Application Process			
Section 3.1			
Section text	-----------------------	No objection	-------------------------------
Figure 3-1	-----------------------	No objection	-------------------------------
Section 3.2			
Section text	-----------------------	No objection	-------------------------------
Section 3.3			
Section text	-----------------------	No objection	-------------------------------
Section 4: Risk Assessment Technical Requirements			
Section 4.1			
Section text	-----------------------	No objection	-------------------------------
Section 4.2			
Section text	-----------------------	No objection	-------------------------------
Section 4.2.1			
Section text	-----------------------	No objection	-------------------------------
Section 4.2.2			
Section text	-----------------------	No objection	-------------------------------
Section 4.2.3			
Section text	-----------------------	No objection	-------------------------------
Section 4.2.4			
Section text	-----------------------	No objection	-------------------------------
Section 4.2.5			

**Table A-1. Staff Position on ASME/ANS RA-S-1.4-2021,
NLWR Standard Introduction, Acronyms and Definitions, and Risk Assessment
Application Process**

Index No.	Issue	Position	Resolution
Section text	-----------------------	No objection	------------------------------
Section 4.2.6			
Section text	-----------------------	No objection	------------------------------
Section 4.2.7			
Section text	-----------------------	No objection	------------------------------

Table A-2. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Plant Operating State Analysis

Index No.	**Issue**	**Position**	**Resolution**
Section 4.3.1			
Section text	-----------------------	No objection	-------------------------------
Section 4.3.1.1			
Section text	-----------------------	No objection	-------------------------------
Table 4.3.1.1-1			
HLR-POS-A through HLR-POS-D	-----------------------	No objection	-------------------------------
Table 4.3.1.1-2			
POS-A1 through POS-A7	-----------------------	No objection	-------------------------------
POS-A8	Operations personnel supporting a licensing application should support efforts to confirm that the selection of POSs correctly represents the as-designed and as-intended-to-operate plant. Without this supporting confirmation from operations personnel, the PRA may exclude potential risk contributors due to a lesser degree of plant representation.	Qualification	For PRAs performed during the pre-operational stage, INTERVIEW knowledgeable engineering **and operations personnel** to confirm that the selection of plant operating states correctly represents the as-designed, and as-intended-to-operate plant.
POS-A9 through POS-A13	-----------------------	No objection	-------------------------------
Table 4.3.1.1-3			
POS-B1	The requirement should assess whether grouping of POSs impacts risk-significant event sequences. Without such an assessment,	Qualification	<u>CC-I</u> GROUP plant evolutions into a set of representative evolutions. ENSURE that (a) the evolutions within a group can be considered

**Table A-2. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Plant Operating State Analysis**

Index No.	Issue	Position	Resolution
	event sequences that might otherwise be considered risk-significant may not be categorized as such.		similar in terms of the set of plant operating states that they contain; (b) the evolutions are bounded by the worst case impact within the group; **(c) the grouping does not impact risk-significant event sequences.**
POS-B2 through POS-B8	----------------------	No objection	---------------------------------
Table 4.3.1.1-4			
POS-C1 through POS-C5	----------------------	No objection	---------------------------------
Table 4.3.1.1-5			
POS-D1 through POS-D3	----------------------	No objection	---------------------------------
Section 4.3.1.2			
Section text	----------------------	No objection	---------------------------------
Section 4.3.1.2.1			
Section text	----------------------	No objection	---------------------------------
Section 4.3.1.2.2			
Section text	----------------------	No objection	---------------------------------
Section 4.3.1.2.3			
Section text	----------------------	No objection	---------------------------------

Table A-3. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Initiating Event Analysis

Index No.	Issue	Position	Resolution
Section 4.3.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.2.1			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.2.1-1			
HLR-IE-A through HLR-IE-D	-----------------------	No objection	---------------------------------
Table 4.3.2.1-2			
IE-A1 through IE-A18	-----------------------	No objection	---------------------------------
Table 4.3.2.1-3			
IE-B1 through IE-B6	-----------------------	No objection	---------------------------------
Table 4.3.2.1-4			
IE-C1 through IE-C8	-----------------------	No objection	---------------------------------
IE-C9	If operator actions are relied upon to detect and correct conditions that may lead to a complicated shutdown, the reliability of those actions should be shown to have an exceedingly low probability of failure (i.e., the collective failure probability for all operator actions in a given event sequence is less than or equal to 10^{-5}) as assessed against human reliability analysis requirements for NLWR PRA endorsed by the NRC.	Clarification	*(b)* either: *(1)* the event has the same impact on the plant as another event that has a much higher frequency per the requirements of SCR-1 or SCR-2 in Table 1.10-1, or *(2)* the event does not require the plant to go to shutdown conditions until sufficient time has expired during which the initiating event conditions, with a high degree of certainty (based on supporting calculations **such that the collective failure probability is less than or equal to 10^{-5} and in conformance with human reliability technical elements**, are detected and corrected (either administratively or automatically) such that a

Table A-3. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Initiating Event Analysis

Index No.	Issue	Position	Resolution
			complicated shutdown does not occur per the requirements of SCR-3 in Table 1.10-1.
IE-C10 through IE-C19	-----------------------	No objection	---------------------------------
Table 4.3.2.1-5			
IE-D1 through IE-D3	-----------------------	No objection	---------------------------------
Section 4.3.2.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.2.2.1			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.2.2.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.2.2.3			
Section text	-----------------------	No objection	---------------------------------

Table A-4. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Event Sequence Analysis

Index No.	Issue	Position	Resolution
Section 4.3.3			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.3.1			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.3.1-1			
HLR-ES-A through HLR-ES-D	-----------------------	No objection	---------------------------------
Table 4.3.3.1-2			
ES-A1 through ES-A15	-----------------------	No objection	---------------------------------
Table 4.3.3.1-3			
ES-B1 through ES-B10	-----------------------	No objection	---------------------------------
Table 4.3.3.1-4			
ES-C1 through ES-C11	-----------------------	No objection	---------------------------------
Table 4.3.3.1-5			
ES-D1 through ES-D3	-----------------------	No objection	---------------------------------
Section 4.3.3.2			
Section 4.3.3.2.1			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.3.2.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.3.2.3			
Section text	-----------------------	No objection	---------------------------------

Table A-5. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Success Criteria Analysis

Index No.	Issue	Position	Resolution
Section 4.3.4			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.4.1			
Section text	-----------------------	No objection	--------------------------------
Table 4.3.4.1-1			
HLR-SC-A through HLR-SC-C	-----------------------	No objection	--------------------------------
Table 4.3.4.1-2			
SC-A1 through SC-A11	-----------------------	No objection	--------------------------------
Table 4.3.4.1-3			
SC-B1 through SC-B10	-----------------------	No objection	--------------------------------
Table 4.3.4.1-4			
SC-C1 through SC-C3	-----------------------	No objection	--------------------------------
Section 4.3.4.2			
Section 4.3.4.2.1			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.4.2.2			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.4.2.3			
Section text	-----------------------	No objection	--------------------------------

Table A-6. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Systems Analysis

Index No.	Issue	Position	Resolution
Section 4.3.5			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.5.1			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.5.1-1			
HLR-SY-A through HLR-SY-C	-----------------------	No objection	---------------------------------
Table 4.3.5.1-2			
SY-A1 through SY-A33	-----------------------	No objection	---------------------------------
Table 4.3.5.1-3			
SY-B1 through SY-B17	-----------------------	No objection	---------------------------------
Table 4.3.5.1-4			
SY-C1 through SY-C3	-----------------------	No objection	---------------------------------
Section 4.3.5.2			
Section 4.3.5.2.1			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.5.2.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.5.2.3			
Section text	-----------------------	No objection	---------------------------------

Table A-7. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Human Reliability Analysis

Index No.	Issue	Position	Resolution
Section 4.3.6			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.6.1			
Section text	Use of the term "unscreened" may imply that an item was previously screened out and subsequently screened back into the PRA. However, this term is interpreted as meaning activities that were included in the PRA for consideration and evaluation (i.e., screened in).	Clarification	...*(c)* human failure events (HFEs) are defined for ~~unscreened~~ activities **modeled in the PRA**;
Table 4.3.6.1-1			
HLR-HR-A through HLR-HR-D	-----------------------	No objection	--------------------------------
HLR-HR-E	Not including errors of commission (EOCs) presumes that EOCs are not important or significant contributors to the NLWR PRA; however, it is currently not known whether that is the case. Though there is significant experience with operating light-water reactors (LWRs) to justify the consensus approach of excluding EOCs from the Level 1/LERF LWR PRA standard, there is	Qualification	A systematic review of relevant available procedures, any past operational events, procedural guidance, and training shall be used to identify the set of post-initiator operator responses required for each of the event sequences**, as well as the well-intended post-initiator operator responses that result in adverse safety impacts**.

Table A-7. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Human Reliability Analysis

Index No.	Issue	Position	Resolution
	little to no operating experience for NLWRs dictating that the same approach can be applied. It is expected that NLWRs will rely less on human actions than operating LWRs, which implies that EOCs would play a more important role in NLWR PRAs. In addition, there is no consensus for LWRs regarding whether EOCs should be addressed in the accident progression analysis (i.e., Level 2 PRA). Likewise, there is currently no basis for excluding EOCs from the mechanistic source term analysis in a NLWR PRA.		
HLR-HR-F through HLR-HR-I	------------------------	No objection	----------------------------------
Table 4.3.6.1-2			
HR-A1 through HR-A10	------------------------	No objection	----------------------------------
Table 4.3.6.1-3			
HR-B1 through HR-B3	------------------------	No objection	----------------------------------
Table 4.3.6.1-4			
HR-C1 through HR-C6	------------------------	No objection	----------------------------------
Table 4.3.6.1-5			

Table A-7. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Human Reliability Analysis

Index No.	Issue	Position	Resolution
HR-D1 through HR-D3	------------------------	No objection	----------------------------------
HR-D4	The detailed human error probability (HEP) assessments for an operating plant would not be realistic due to not including the information listed in the CC-II of HR-D4. Using the phrase “when available,” with respect to an operating plant, may result in the CC-II of HR-D4 being interpreted as not applicable due to the information not being available. For an operating plant, this information should be available.	Clarification	CC-II For each detailed HEP assessment, INCLUDE in the evaluation process the following plant- or design-specific relevant information ~~when available~~:
HR-D5	------------------------	No objection	----------------------------------
HR-D6	The recovery of pre-initiator errors assessment would be incomplete due to not using all the information listed in HR-D6. By using the phrase “if available,” HR-D6 may be interpreted as not applicable due to the information not being available. For an operating plant, all the information listed in HR-D6 should be available.	Clarification	For operating plants, if recovery of pre-initiator errors is credited, USE the following information~~, if available,~~ to assess the potential...
HR-D7 through HR-D10	------------------------	No objection	----------------------------------

Table A-7. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Human Reliability Analysis

Index No.	Issue	Position	Resolution
Table 4.3.6.1-6			
HR-E1 through HR-E3	-----------------------	No objection	---------------------------------
HR-E4	Not including errors of commission (EOCs) presumes that EOCs are not important or significant contributors to NLWR PRA; however, it is currently not known whether that is the case. Though there is significant experience with operating light-water reactors to justify the consensus approach of excluding errors of commission (EOCs) from the Level 1/LERF LWR PRA standard, there is little to no operating experience for NLWRs justifying that the same approach can be applied. It is expected that NLWRs will rely less on human actions than operating LWRs, which implies that EOCs would play a more important role in NLWR PRAs. Additionally, there is no consensus for LWRs regarding whether EOCs should be addressed in the	Qualification	*Add the following to item HR-E4*: "**(c) those well-intended actions performed by control room staff that disable a system, subsystem, or component needed in an event scenario.**"

Table A-7. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Human Reliability Analysis

Index No.	**Issue**	**Position**	**Resolution**
	accident sequence progression analysis (i.e., Level 2 PRA). Likewise, there is currently no basis for excluding EOCs from the mechanistic source term analysis in a NLWR PRA.		
HR-E5 through HR-E9	------------------------	No objection	----------------------------------
Table 4.3.6.1-7			
HR-F1 through HR-F5	------------------------	No objection	----------------------------------
Table 4.3.6.1-8			
HR-G1	Some HRA methods include a feasibility step during the qualitative analysis and other HRA methods do not include the step. Since it is up to the NLWR PRA developer to choose which HRA method to use, not all HRA methods include this step, and feasibility assessment is a continuous step in the HRA process, it is important to assess the feasibility of a human action (even during quantification). Further, ESQ-C7 requires that human actions be feasible in order to use their respective HEPs in event sequence quantification. Since the NLWR PRA	Qualification	CC-I **ASSESS the feasibility of the HFEs before assigning the final HEPs using the criteria in HR-H2. For HFEs determined to be feasible,** USE conservative estimates for the HEPs of the HFEs in the event sequences-that survive initial quantification. CC-II **ASSESS the feasibility of the HFEs before assigning the final HEPs using the criteria in HR-H2. For HFEs determined to be feasible,** PERFORM detailed analyses for estimation of HEPs for risk-significant HFEs. For the HEPs of HFEs that are not risk-significant, ENSURE the requirement for CC-I is met.

Table A-7. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Human Reliability Analysis

Index No.	Issue	Position	Resolution
	standard does not explicitly require a feasibility assessment, this step in the HRA process may be missed, which would cause the PRA model to underestimate the risk from that human action.		
HR-G2 and HR-G3	-----------------------	No objection	---------------------------------
HR-G4	Communication is an important performance-shaping factor for actions taken outside the main control room (e.g., in fire scenarios) and that require complex coordination, such as the use of FLEX equipment. As such, all aspects of communication should be evaluated when estimating HEPs.	Clarification	<u>CC-II</u> **(p) communication among personnel on the same team and on different teams.**
HR-G5 through HR-G16	-----------------------	No objection	---------------------------------
Table 4.3.6.1-9			
HR-H1	-----------------------	No objection	---------------------------------
HR-H2	Having a limited set of feasibility criteria and constraining their applicability only to recovery actions (i.e., a special case of post-initiator actions) may result in an incomplete feasibility assessment and	Qualification	*Add the following two feasibility criteria:* **(f) there is a sufficient plan for command and control; and** **(g) there is a sufficient plan for communications.**

Table A-7. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Human Reliability Analysis

Index No.	Issue	Position	Resolution
	assigning an HEP of less than 1.0 when the assigned value should be 1.0 (see the "Issue" column for HR-G1). Ensuring that there is sufficient command and control, and communications is important because NLWRs lack the operating experience and the NLWR PRA standard scope includes all radiological sources, all hazards, all POSs, and all levels of analysis.		
HR-H3 through HR-H6	-----------------------	No objection	--------------------------------
Table 4.3.6.1-10			
HR-I1 through HR-I3	-----------------------	No objection	--------------------------------
Section 4.3.6.2			
Section 4.3.6.2.1			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.6.2.2			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.6.2.3			
Section text	-----------------------	No objection	--------------------------------

Table A-8. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Data Analysis

Index No.	Issue	Position	Resolution
Section 4.3.7			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.7.1			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.7.1-1			
HLR-DA through HLR-DA-E	-----------------------	No objection	---------------------------------
Table 4.3.7.1-2			
DA-A1 through DA-A6	-----------------------	No objection	---------------------------------
Table 4.3.7.1-3			
DA-B1 through DA-B2	-----------------------	No objection	---------------------------------
Table 4.3.7.1-4			
DA-C1 through DA-C19	-----------------------	No objection	---------------------------------
DA-C20	When crediting equipment repair, generic industry data should not be used in cases where plant-specific data is available. Crediting generic industry data may underestimate risk from a failure to repair event(s), which can mask the importance of other risk contributors.	Qualification	…IDENTIFY instances of plant-specific experience ~~or~~ **and, when that is insufficient to estimate failure to repair consistent with DA-D10,** applicable industry experience and for each repair, COLLECT…
DA-C21 through DA-C26	-----------------------	No objection	---------------------------------
Table 4.3.7.1-5			
DA-D1 through DA-D10	-----------------------	No objection	---------------------------------
Table 4.3.7.1-6			

**Table A-8. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Data Analysis**

Index No.	**Issue**	**Position**	**Resolution**
DA-E1 throughDA-E3	-----------------------	No objection	---------------------------------
Section 4.3.7.2			
Section 4.3.7.2.1			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.7.2.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.7.2.3			
Section text	-----------------------	No objection	---------------------------------

**Table A-9. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Internal Flood Analysis**

Index No.	Issue	Position	Resolution
Section 4.3.8			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.8.1			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.8.1-1			
HLR-FLPP-A through HLR-FLPP-C	-----------------------	No objection	---------------------------------
Table 4.3.8.1-2			
FLPP-A1	-----------------------	No objection	---------------------------------
Table 4.3.8.1-3			
FLPP-B1 through FLPP-B5	-----------------------	No objection	---------------------------------
FLPP-B6	A walkdown should be performed for post-operational plants as part of an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation that involves something less than a walkdown for post-operational plant may result in a mischaracterization the plant's as-built design in the PRA.	Clarification	EVALUATE the Internal Flood Plant Partitioning for the as-built, as-operated or as-designed, as-intended-to-operate plant conditions via **walkdowns or, for PRAs performed during the preoperational phase,** investigation(s) depending…
FLPP-B7 through FLPP-B8	-----------------------	No objection	---------------------------------
Table 4.3.8.1-4			
FLPP-C1	A walkdown should be performed for post-operational plants and should not	Clarification	… *(b)* the general nature and key or unique features of the

Table A-9. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Internal Flood Analysis

Index No.	Issue	Position	Resolution
	be replaced with an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation instead of a walkdown when a walkdown is possible may result in a mischaracterization the plant response in the PRA.		partitioning elements that define each flood area; *(c)* any **walkdowns or, for PRAs performed during the preoperational phase,** investigation(s) performed in support of the plant partitioning; …
FLPP-C2 through FLPP-C3	-----------------------	No objection	--------------------------------
Section 4.3.8.2			
Section text	-----------------------	No objection	--------------------------------
Table 4.3.8.2-1			
HLR-FLSO-A and HLR-FLSO-B	-----------------------	No objection	--------------------------------
Table 4.3.8.2-2			
FLSO-A1	-----------------------	No objection	--------------------------------
FLSO-A2	Fluid sources other than water and steam can produce flooding effects that are unlike water and steam and may result in risk-significant contributors. A lack of consideration of other fluid sources besides water may exclude significant sources of flooding and underestimate risk.	Clarification	IDENTIFY the potential flood sources that include water, steam, **and other liquids (e.g., lubricating oil, fuel oil)**.

Table A-9. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Internal Flood Analysis

Index No.	Issue	Position	Resolution
FLSO-A3 through FLSO-A6	-----------------------	No objection	---------------------------------
FLSO-A7	A walkdown should be performed for post-operational plants and should not be replaced with an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation instead of a walkdown when a walkdown is possible may result in a mischaracterization the plant response in the PRA.	Clarification	… CONFIRM the accuracy of information collected from plant information sources for the as-designed, or as-built, or as-operated and as-intended-to-operate plant conditions via **walkdowns or, for PRAs performed during the preoperational phase,** investigation(s)…
FLSO-A8 through FLSO-A9	-----------------------	No objection	---------------------------------
Table 4.3.8.2-3			
FLSO-B1 through FLSO-B3	-----------------------	No objection	---------------------------------
Section 4.3.8.3			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.8.3-1			
HLR-FLSN-A and HLR-FLSN-B	-----------------------	No objection	---------------------------------
Table 4.3.8.3-2			
FLSN-A1 through FLSN-A18	-----------------------	No objection	---------------------------------
FLSN-A19	A walkdown should be performed for post-operational plants and should not	Clarification	EVALUATE the accuracy of information collected from plant information sources for the as-designed, or as-built, and

Table A-9. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Internal Flood Analysis

Index No.	Issue	Position	Resolution
	be replaced with an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation instead of a walkdown when a walkdown is possible may result in a mischaracterization the plant response in the PRA.		as-operated or as-intended-to-operate plant conditions via **walkdowns or, for PRAs performed during the preoperational phase,** investigation(s) depending…
FLSN-A20 through FLSN-A21	-----------------------	No objection	---------------------------------
Table 4.3.8.3-3			
FLSN-B1	A walkdown should be performed for post-operational plants and should not be replaced with an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation instead of a walkdown when a walkdown is possible may result in a mischaracterization the plant response in the PRA.	Clarification	… *(f)* calculations or other analyses used to support or refine the flooding evaluation; *(g)* **walkdowns or, for PRAs performed during the preoperational phase,** investigation(s) performed; …
FLSN-B2 through FLSN-B3	-----------------------	No objection	---------------------------------
Section 4.3.8.4			

**Table A-9. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Internal Flood Analysis**

Index No.	Issue	Position	Resolution
Section text	-----------------------	No objection	---------------------------------
Table 4.3.8.4-1			
HR-FLEV-A through HR-FLEV-C	-----------------------	No objection	---------------------------------
Table 4.3.8.4.2			
FLEV-A1 through FLEV-A4	-----------------------	No objection	---------------------------------
Table 4.3.8.4.3			
FLEV-B1 through FLEV-B7	-----------------------	No objection	---------------------------------
Table 4.3.8.4.4			
FLEV-C1 through FLEV-C3	-----------------------	No objection	---------------------------------
Section 4.3.8.5			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.8.5-1			
HLR-FLPR-A through HLR-FLPR-C	-----------------------	No objection	---------------------------------
Table 4.3.8.5-2			
FLPR-A1 through FLPR-A3	-----------------------	No objection	---------------------------------
Table 4.3.8.5-3			
FLPR-B1 through FLPR-B10	-----------------------	No objection	---------------------------------
Table 4.3.8.5-4			
FLPR-C1 through FLPR-C3	-----------------------	No objection	---------------------------------
Section 4.3.8.6			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.8.6-1			
HLR-FLHR-A through HLR-FLHR-E	-----------------------	No objection	---------------------------------

Table A-9. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Internal Flood Analysis

Index No.	Issue	Position	Resolution
Table 4.3.8.6-2			
FLHR-A1 through FLHR-A2	-----------------------	No objection	---------------------------------
Table 4.3.8.6-3			
FLHR-B1 through FLHR-B3	-----------------------	No objection	---------------------------------
Table 4.3.8.6-4			
FLHR-C1	-----------------------	No objection	---------------------------------
Table 4.3.8.6-5			
FLHR-D1 through FLHR-D3	-----------------------	No objection	---------------------------------
Table 4.3.8.6-6			
FLHR-E1 through FLHR-E3	-----------------------	No objection	---------------------------------
Section 4.3.8.7			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.8.7-1			
HLR-FLESQ-A through HLR-FLESQ-F	-------------------	No objection	---------------------------------
Table 4.3.8.7-2			
FLESQ-A1 through FLESQ-A5	-----------------------	No objection	---------------------------------
FLESQ-A6	Walkdowns should be performed for post-operational plants and should not be replaced with an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible.	Clarification	COLLECT inputs to the following analyses, which support quantifications of flood-induced event sequences, from plant or design information sources, as applicable, or via **walkdowns or, for PRAs performed during the preoperational phase,** investigation(s):…

Table A-9. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Internal Flood Analysis

Index No.	Issue	Position	Resolution
	Performing an investigation instead of a walkdown when a walkdown is possible may result in a mischaracterization the plant response in the PRA.		
FLESQ-A7 through FLESQ-A8	-----------------------	No objection	---------------------------------
Table 4.3.8.7-3			
FLESQ-B1	-----------------------	No objection	---------------------------------
Table 4.3.8.7-4			
FLESQ-C1	-----------------------	No objection	---------------------------------
Table 4.3.8.7-5			
FLESQ-D1	-----------------------	No objection	---------------------------------
Table 4.3.8.7-6			
FLESQ-E1 through FLESQ-E2	-----------------------	No objection	---------------------------------
Table 4.3.8.7-7			
FLESQ-F1 through FLESQ-F5	-----------------------	No objection	---------------------------------
Section 4.3.8.8			
Section 4.3.8.8.1			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.8.8.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.8.8.3			
Section 4.3.8.8.3.1			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.8.8.3.2			
Section text	-----------------------	No objection	---------------------------------

Table A-9. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Internal Flood Analysis

Index No.	Issue	Position	Resolution
Section 4.3.8.8.3.3			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.8.8.3.4			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.8.8.3.5			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.8.8.3.6			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.8.8.3.7			
Section text	-----------------------	No objection	--------------------------------

Table A-10. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Internal Fire PRA

Index No.	Issue	Position	Resolution
Section 4.3.9			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.9.1			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.9.1-1			
HLR-FPP-A through HLR-FPP-C	-----------------------	No objection	---------------------------------
Table 4.3.9.1-2			
FPP-A1	-----------------------	No objection	---------------------------------
Table 4.3.9.1-3			
FPP-B1 through FPP-B8	-----------------------	No objection	---------------------------------
Table 4.3.9.1-4			
FPP-C1 through FPP-C3	-----------------------	No objection	---------------------------------
Section 4.3.9.2			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.9.2-1			
HLR-FES-A through HLR-FES-D	-----------------------	No objection	---------------------------------
Table 4.3.9.2-2			
FES-A1 through FES-A7	-----------------------	No objection	---------------------------------
Table 4.3.9.2-3			
FES-B1 through FES-B3	-----------------------	No objection	---------------------------------
Table 4.3.9.2-4			
FES-C1 through FES-C3	-----------------------	No objection	---------------------------------
Table 4.3.9.2-5			

Table A-10. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Internal Fire PRA

Index No.	Issue	Position	Resolution
FES-D1 through FES-D3	------------------------	No objection	----------------------------------
Section 4.3.9.3			
Section text	------------------------	No objection	----------------------------------
Table 4.3.9.3-1			
HLR-FCS-A through HLR-FCS-C	------------------------	No objection	----------------------------------
Table 4.3.9.3-2			
FCS-A1 through FCS-A4	------------------------	No objection	----------------------------------
Table 4.3.9.3-3			
FCS-B1 through FCS-B3	------------------------	No objection	----------------------------------
Table 4.3.9.3-4			
FCS-C1 through FCS-C3	------------------------	No objection	----------------------------------
Section 4.3.9.4			
Section text	------------------------	No objection	----------------------------------
Table 4.3.9.4-1			
HR-FQLS-A through HR-FQLS-B	------------------------	No objection	----------------------------------
Table 4.3.9.4-2			
FQLS-A1 through FQLS-A6	------------------------	No objection	----------------------------------
Table 4.3.9.4-3			
FQLS-B1 through FQLS-B3	------------------------	No objection	----------------------------------
Section 4.3.9.5			
Section text	------------------------	No objection	----------------------------------
Table 4.3.9.5-1			

Table A-10. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Internal Fire PRA

Index No.	Issue	Position	Resolution
HLR-FPRM-A through HLR-FPRM-C	-----------------------	No objection	--------------------------------
Table 4.3.9.5-2			
FPRM-A1 through FPRM-A3	-----------------------	No objection	--------------------------------
Table 4.3.9.5-3			
FPRM-B1 through FPRM-B17	-----------------------	No objection	--------------------------------
Table 4.3.9.5-4			
FPRM-C1 through FPRM-C4	-----------------------	No objection	--------------------------------
Section 4.3.9.6			
Section text	-----------------------	No objection	--------------------------------
Table 4.3.9.6-1			
HLR-FSS-A through HLR-FSS-H	-----------------------	No objection	--------------------------------
Table 4.3.9.6-2			
FSS-A1 through FSS-A4	-----------------------	No objection	--------------------------------
Table 4.3.9.6-3			
FSS-B1 through FSS-B2	-----------------------	No objection	--------------------------------
Table 4.3.9.6-4			
FSS-C1 through FSS-C7	-----------------------	No objection	--------------------------------
Table 4.3.9.6-5			
FSS-D1 through FSS-D11	-----------------------	No objection	--------------------------------
Table 4.3.9.6-6			
FSS-E1 through FSS-E5	-----------------------	No objection	--------------------------------

Table A-10. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Internal Fire PRA

Index No.	Issue	Position	Resolution
Table 4.3.9.6-7			
FSS-F1 through FSS-F2	-----------------------	No objection	---------------------------------
Table 4.3.9.6-8			
FSS-G1 through FSS-G9	-----------------------	No objection	---------------------------------
Table 4.3.9.6-9			
FSS-H1 through FSS-H4	-----------------------	No objection	---------------------------------
Section 4.3.9.7			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.9.7-1			
HLR-FIGN-A through HLR-FIGN-B	-----------------------	No objection	---------------------------------
Table 4.3.9.7-2			
FIGN-A1 through FIGN-A12	-----------------------	No objection	---------------------------------
Table 4.3.9.7-3			
FIGN-B1 through FIGN-B3	-----------------------	No objection	---------------------------------
Section 4.3.9.8			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.9.8.1			
HLR-FCF-A through HLR-FCF-B	-----------------------	No objection	---------------------------------
Table 4.3.9.8-2			
FCF-A1 through FCF-A4	-----------------------	No objection	---------------------------------
Table 4.3.9.8-3			
FCF-B1 through FCF-B3	-----------------------	No objection	---------------------------------
Section 4.3.9.9			

**Table A-10. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Internal Fire PRA**

Index No.	Issue	Position	Resolution
Section text	------------------------	No objection	----------------------------------
Table 4.3.9.9-1			
HLR-FHR-A through HLR-FHR-E	------------------------	No objection	----------------------------------
Table 4.3.9.9-2			
FHR-A1 through FHR-A3	------------------------	No objection	----------------------------------
Table 4.3.9.9-3			
FHR-B1 through FHR-B2	------------------------	No objection	----------------------------------
Table 4.3.9.9-4			
FHR-C1	------------------------	No objection	----------------------------------
Table 4.3.9.9-5			
FHR-D1 through FHR-D3	------------------------	No objection	----------------------------------
Table 4.3.9.9-6			
FHR-E1 through FHR-E3	------------------------	No objection	----------------------------------
Section 4.3.9.10			
Section text	------------------------	No objection	----------------------------------
Table 4.3.9.10-1			
HLR-FESQ-A through HLR-FESQ-F	------------------------	No objection	----------------------------------
Table 4.3.9.10-2			
FESQ-A1 through FESQ-A5	------------------------	No objection	----------------------------------
Table 4.3.9.10-3			
FESQ-B1	------------------------	No objection	----------------------------------
Table 4.3.9.10-4			
FESQ-C1	------------------------	No objection	----------------------------------
Table 4.3.9.10-5			

Table A-10. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Internal Fire PRA

Index No.	Issue	Position	Resolution
FESQ-D1 through FESQ-D3	-----------------------	No objection	---------------------------------
Table 4.3.9.10-6			
FESQ-E1 through FESQ-E2	-----------------------	No objection	---------------------------------
Table 4.3.9.10-7			
FESQ-F1 through FESQ-F4	-----------------------	No objection	---------------------------------
Section 4.3.9.11			
Section 4.3.9.11.1			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.9.11.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.9.11.3			
Section 4.3.9.11.3.1			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.9.11.3.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.9.11.3.3			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.9.11.3.4			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.9.11.3.5			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.9.11.3.6			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.9.11.3.7			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.9.11.3.8			

**Table A-10. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Internal Fire PRA**

Index No.	Issue	Position	Resolution
Section text	------------------------	No objection	----------------------------------
Section 4.3.9.11.3.9			
Section text	------------------------	No objection	----------------------------------
Section 4.3.9.11.3.10			
Section text	------------------------	No objection	----------------------------------

**Table A-11. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Seismic PRA**

Index No.	Issue	Position	Resolution
Section 4.3.10			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.10.1			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.10.1-1			
HLR-SHA-A through HLR-SHA-I	-----------------------	No objection	---------------------------------
Table 4.3.10.1-2			
SHA-A1 through SHA-A7	-----------------------	No objection	---------------------------------
Table 4.3.10.1-3			
SHA-B1 through SHA-B5	-----------------------	No objection	---------------------------------
Table 4.3.10.1-4			
SHA-C1 through SHA-C5	-----------------------	No objection	---------------------------------
Table 4.3.10.1-5			
SHA-D1 and SHA-D2	-----------------------	No objection	---------------------------------
SHA-D3	Ground motion prediction equations with alternative distance and magnitude scaling behaviors accounts for an important source of uncertainty in the ground motion characterization model. If only a range of amplitudes is considered in the ground motion characterization model without alternative distance and magnitude scaling, the	Clarification	ENSURE that uncertainties are included in the model such that the aggregate of predicted ground motions captures the range of ground motions that can occur at a site **as well as alternative magnitude and distance scaling** in accordance with the level of analysis identified for the SRs of HLR-SHA-A and the data and information identified in the SRs of HLR-SHA-B. …

Table A-11. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Seismic PRA

Index No.	Issue	Position	Resolution
	uncertainty in the seismic hazard may be underestimated.		
SHA-D4	-----------------------	No objection	---------------------------------
Table 4.3.10.1-6			
SHA-E1 through SHA-E6	-----------------------	No objection	---------------------------------
Table 4.3.10.1-7			
SHA-F1 through SHA-F4	-----------------------	No objection	---------------------------------
Table 4.3.10.1-8			
SHA-G1 through SHA-G2	-----------------------	No objection	---------------------------------
Table 4.3.10.1-9			
SHA-H1 through SHA-H4	-----------------------	No objection	---------------------------------
Table 4.3.10.1-10			
SHA-I1 through SHA-I3	-----------------------	No objection	---------------------------------
Section 4.3.10.2			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.10.2-1			
HLR-SFR-A through HLR-SFR-F	-----------------------	No objection	---------------------------------
Table 4.3.10.2-2			
SFR-A1 and SFR-A2	-----------------------	No objection	---------------------------------
Table 4.3.10.2-3			
SFR-B1 through SFR-B6	-----------------------	No objection	---------------------------------
Table 4.3.10.2-4			
SFR-C1	The inherently rugged component groups are based on	Clarification	SPECIFY the basis for screening of inherently rugged

Table A-11. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Seismic PRA

Index No.	Issue	Position	Resolution
	earthquake experience and data from operating light-water reactors. Nonlight-water reactors can have SSCs that differ in form and design from light-water SSCs. This may result in either these SSCs not being covered by the current inherently rugged component groups or demonstrating that these SSCs are similar to a particular current inherently rugged component group. Further, the bounding site or range of sites can have a seismic hazard exceeding that used to develop the current inherently rugged component list. Therefore, use of the inherently rugged component groups based on LWRs needs justification and the basis for screening of inherently rugged components needs to be specified. Inappropriate screening of inherently rugged components may result in their exclusion from a seismic PRA and/or in an underestimation of risk.		components **justifying the applicability to the plant.**

Table A-11. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Seismic PRA

Index No.	**Issue**	**Position**	**Resolution**
SFR-C2	The methodologies and approaches for determining the fragility thresholds is based on operating light-water reactor seismic risk and seismic PRA experience (e.g., generic lognormal uncertainty parameters dependent on type or location of SSC, relative contribution to total seismic risk of assuming that a failure leads directly to core damage).These state-of-practice methodologies and approaches may not be applicable to nonlight-water designs or their applicability may need a design-specific technical basis. Further, the same threshold may not be applicable to a range of sites and, for such cases, use of a single threshold would need justification. Inappropriate fragility thresholds may result in inadvertent screening out of components from inclusion in an SPRA and/or an underestimation of risk.	Clarification	SPECIFY the basis and methodologies established for achieving the fragility thresholds defined in Requirement SPR-B5 **justifying the applicability to the plant and site or range of sites identified in SHA-A1.**

Table A-11. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Seismic PRA

Index No.	Issue	Position	Resolution
Table 4.3.10.2-5			
SFR-D1	------------------------	No objection	---------------------------------
SFR-D2	A walkdown should be performed for post-operational plants as part of an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation that involves something less than a walkdown for post-operational plant may result in a mischaracterization the plant's as-built design in the PRA.	Clarification	EVALUATE the seismic capacity of the as-designed, or as-built, or as-operated or as-intended-to-operate plant conditions via **walkdowns or, for PRAs performed during the preoperational phase,** investigation(s).
SFR-D3 and SFR-D4	------------------------	No objection	---------------------------------
SFR-D5	Pathways for operators to access SSCs or controls outside the control room and potential pathway unavailability should be included in the walkdown evaluation of potential functional and structural failure mechanisms, equipment anchorage, and support load paths to ensure that these unavailabilities are accounted for in the feasibility evaluation of operator actions considered for the SPRA. The	Clarification	EVALUATE potential functional and structural failure mechanisms, equipment anchorage, ~~and~~ support load path~~.~~**, and pathways necessary for performing required ex-control room actions.**

Table A-11. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Seismic PRA

Index No.	Issue	Position	Resolution
	walkdown information is used not only by fragility analysts but also by plant response analysts. Therefore, such information will inform the plant response analysts about the feasibility of certain operator actions. If this evaluation is not performed during the walkdown, the PRA may include operator actions that would be infeasible due to a necessary path being unavailable and therefore, impact the plant response characteristics and insights.		
SFR-D6 through SFR-D8	-----------------------	No objection	---------------------------------
Table 4.3.10.2-6			
SFR-E1 and SFR-E2	-----------------------	No objection	---------------------------------
SFR-E3	The applicability of generic data to a given SSC should be justified. This is because of the potential for SSCs in NLWR designs that may not correspond to established SSC categories and/or may not be in common SSC locations in the plant. When estimating fragilities for PRAs developed for the pre-	Clarification	<u>CC-I</u> ESTIMATE conservative seismic fragilities for the failure mechanisms of interest identified in Requirement SFR-E1 using plant-specific data, or JUSTIFY 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 **applicable to the SSC and** appropriate for

Table A-11. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Seismic PRA

Index No.	Issue	Position	Resolution
	operational phase, demonstrating that generic or conservative assumptions are bounding for the range of anticipated sites helps ensure that these fragilities can reasonably reflect the as-designed or as-to-be-built plant. If such assumptions cannot be shown to be bounding for the range of anticipated sites, the results may not appropriately reflect the expected plant response.		the plant. <u>CC-II</u> CALCULATE realistic seismic fragilities for the failure mechanisms of interest identified in Requirement SFR-E1 using plant-specific data, or JUSTIFY (e.g., through the calculation of integrated risk metrics defined in Requirement RI-B3) 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 as being **applicable to the SSC and** appropriate for the plant or by showing no masking or differences in insights.
SFR-E4	The applicability of generic data to a given SSC should be justified. This is because of the potential for SSCs in NLWR designs that may not correspond to established SSC categories and/or may not be in common SSC locations in the plant. When estimating fragilities for PRAs developed for the pre-operational phase, demonstrating that generic or conservative assumptions are bounding for the range of anticipated sites helps ensure that these fragilities can	Clarification	<u>CC-I</u> ESTIMATE contact chatter seismic fragilities for relays and other similar devices that affect SSCs identified in the Systems Analysis and **JUSTIFY the use of generic fragility data or conservative assumptions as being applicable and appropriate for the plant**. <u>CC-II</u> CALCULATE contact chatter seismic fragilities using plant-specific data or JUSTIFY the **applicability and** use of generic fragility data for relays and other similar devices that affect risk-significant SSCs and are identified in the Systems Analysis.

Table A-11. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Seismic PRA

Index No.	Issue	Position	Resolution
	reasonably reflect the as-designed or as-to-be-built plant. If such assumptions cannot be shown to be bounding for the range of anticipated sites, the results may not appropriately reflect the expected plant response.		
SFR-E5	The applicability of generic data to a given SSC should be justified. This is because of the potential for SSCs in NLWR designs that may not correspond to established SSC categories and/or may not be in common SSC locations in the plant. When estimating fragilities for PRAs developed for the pre-operational phase, demonstrating that generic or conservative assumptions are bounding for the range of anticipated sites helps ensure that these fragilities can reasonably reflect the as-designed or as-to-be-built plant. If such assumptions cannot be shown to be bounding for the range of anticipated sites, the results may not appropriately reflect the expected	Clarification	<u>CC-I</u> ESTIMATE seismic fragilities for credible seismic-induced flood sources (see Requirement SFR-D6) and seismic-induced fire ignition sources (see Requirement SFR-D7) **and JUSTIFY the use of generic fragility data or conservative assumptions as being applicable and appropriate for the plant**. <u>CC-II</u> CALCULATE seismic fragilities using plant-specific data or JUSTIFY the **applicability and** use of generic fragility data for credible seismic-induced flood sources (see Requirement SFR-D6) and seismic-induced fire ignition sources (see Requirement SFR-D7) that are risk-significant contributors.

Table A-11. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Seismic PRA

Index No.	Issue	Position	Resolution
	plant response.		
SFR-E6	-----------------------	No objection	---------------------------------
SFR-E7	The sources of uncertainty, related assumptions, and reasonable alternatives for the seismic fragility analysis for pre-operational stage PRAs are not coupled to SR SPR-E8, which determines the impact of the sources of uncertainty on event sequence families and the resulting impact on decisions using the PRA. Merely identifying the sources of uncertainties for pre-operational phase PRAs does not provide sufficient information to determine their impact on the event sequence families and the decisions therefrom. The sources of uncertainty in pre-operational stage PRAs are expected to be different from those during the operational stages.	Qualification	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence the Seismic Fragility Analysis **in a manner that supports Requirement SPR-E8**.
Table 4.3.10.2.7			
SFR-F1 through SFR-F3	-----------------------	No objection	---------------------------------
Section 4.3.10.3			
Section text	-----------------------	No objection	---------------------------------

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Technical Requirements for Seismic PRA

Index No.	Issue	Position	Resolution
Table 4.3.10.3-1			
HLR-SPR-A through HLR-SPR-F	-----------------------	No objection	---------------------------------
Table 4.3.10.3-2			
SPR-A1 through SPR-A4	-----------------------	No objection	---------------------------------
Table 4.3.10.3-3			
SPR-B1 through SPR-B13	-----------------------	No objection	---------------------------------
Table 4.3.10.3-4			
SPR-C1 through SPR-C6	-----------------------	No objection	---------------------------------
Table 4.3.10.3-5			
SPR-D1 through SPR-D4	-----------------------	No objection	---------------------------------
SPR-D5	The feasibility of operator actions in seismic PRA should be assessed. Further, ESQ-C7 requires that human actions be feasible to use their HEPs in event sequence quantification. The CC-II requirement incorrectly states that CC-I of HLR-HR-G should be satisfied.	Qualification	CC-I When addressing **feasibility,** influencing factors, and the timing considerations covered in Requirements **HR-G1,** HR-G4, HR-G6, and HR-G8, INCLUDE... CC-II For developing HEPs, SATISFY the Capability Category ~~I~~ **II** SRs of HLR-HR-G, except where the requirements are not applicable, taking into account relevant seismic related effects on control room and ex-control room post-initiator actions. When addressing **feasibility** influencing factors and the timing considerations covered in Requirements **HR-G1**, HR-G4, HR-G6, and HR-G8, INCLUDE the effect of the seismic hazard on the control

Table A-11. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Seismic PRA

Index No.	Issue	Position	Resolution
			room and ex-control room human actions.
Table 4.3.10.3-6			
SPR-E1 through SPR-E7	-----------------------	No objection	---------------------------------
SPR-E8	This SR evaluates the sources of uncertainty, related assumptions, and reasonable alternatives for the seismic fragility analysis. This SR should be coupled with SFR-E6 because it is important to determine and disposition and impact of these sources of uncertainty for pre-operational stage PRAs. Not identifying and dispositioning the impacts of the identified sources of model uncertainty, related assumptions, and reasonable alternatives for pre-operational stage PRAs can negatively influence the decisions made using these PRAs.	Clarification	SATISFY Requirement ESQ-E1 with the additional assumptions identified by each seismic technical sub-element in Requirement SHA-F3, fragility analysis, Requirement SFR-E6**, SFR-E7**, or both and system modeling, Requirement SPR-E6, **SPR-E7** or both.
Table 4.3.10.3-7			
SPR-F1 through SPR-F5	-----------------------	No objection	---------------------------------
Section 4.3.10.4			
Section 4.3.10.4.1			
Section text	-----------------------	No objection	---------------------------------

Table A-11. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Seismic PRA

Index No.	Issue	Position	Resolution
Section 4.3.10.4.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.10.4.3			
Section 4.3.10.4.3.1			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.10.4.3.2			
Section 4.3.10.4.3.2.1			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.10.4.3.2.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.10.4.3.2.3			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.10.4.3.3.1			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.10.4.3.3.2			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.10.4.3.3.3			
Section text	-----------------------	No objection	---------------------------------

Table A-12. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Hazard Screening Analysis

Index No.	Issue	Position	Resolution
Section 4.3.11			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.11.1			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.11.1-1			
HLR-HS-A through HLR-HS-E	-----------------------	No objection	---------------------------------
Table 4.3.11.1-2			
HS-A1 through HS-A2	-----------------------	No objection	---------------------------------
HS-A3	Plant -specific hazards should be identified as part of the identification of hazards or hazard groups. If plant-specific hazards or hazard groups are not identified, the PRA model may exclude significant risk contributors, underestimate risk, or both.	Clarification	IDENTIFY site-**, plant-, or** ~~and~~ design-unique hazards and hazard groups not already identified in Requirement HS-A2.
HS-A4	-----------------------	No objection	---------------------------------
Table 4.3.11.1-3			
HS-B1	-----------------------	No objection	---------------------------------
HS-B2	Limiting reviews of information about regional-, industrial-, governmental-, and plant-funded evaluations for each hazard only to operating plants potentially excludes relevant hazard information for plants	Clarification	~~For PRAs performed on operating plants,~~ REVEW information about regional-, industrial-, governmental-, and plant-funded evaluations for each hazard, if available.

Table A-12. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Hazard Screening Analysis

Index No.	Issue	Position	Resolution
	in the pre-operational stages of the plant life cycle. These types of reviews should be performed for any stage of the plant life cycle.		
HS-B3 through HS-B4	------------------------	No objection	----------------------------------
HS-B5	When assessing whether a hazard or hazard group cannot physically impact the plant or plant operations, it is necessary to consider a range of magnitudes and frequencies. Without such consideration, lower frequency events with a higher magnitude that might otherwise prove important to the risk results may be excluded. Additionally, the values in RI-A5 referenced in item (f) are presented as reporting values, not screening values. Using the reporting values as screening values could be too permissive in excluding contributors from the PRA as screening using a consequence criterion may not be effectively equivalent to screening using a frequency criterion. Additionally, this	Qualification	USE SCR-3 in Table 1.10-1 when **qualitatively** screening out a hazard or hazard group by showing that either: (a) the hazard or hazard group cannot physically impact the plant or plant operations (e.g., it cannot occur close enough to the plant to affect it)**, taking into account the range of magnitudes and frequencies of the hazard or hazard group**; (b) the hazard or hazard group does not result in a plant trip (manual or automatic) or require a plant shutdown; (c) the hazard or hazard group is included in the definition of another hazard; (d) the hazard or hazard group could not result in worse effects to the plant as another hazard that has a significantly higher frequency; (e) the hazard or hazard group is slow in developing and there is demonstrably sufficient time to eliminate the source of the threat or to provide an adequate response; ~~(f) the hazard or hazard group cannot produce a consequence above the value set in RI-A5.~~

Table A-12. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Hazard Screening Analysis

Index No.	Issue	Position	Resolution
	requirement is effectively for qualitative screening, as per SCR-3 in Table 1.10-1. Because item (f) is a quantitative criterion, it should not be included in the list.		
HS-B6 through HS-B7	-----------------------	No objection	---------------------------------
Table 4.3.11.1-4			
HS-C1 through HS-C14	-----------------------	No objection	---------------------------------
Table 4.3.11.1-5			
HS-D1	A walkdown should be performed for post-operational plants as part of an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation that involves something less than a walkdown for a post-operational plant may result in a mischaracterization of the plant's as-built design in the PRA.	Clarification	CONFIRM that the basis for the screening out of a hazard or hazard group represents either the as-built, as-operated or as-designed, as-intended-to-operate plant conditions **via a walkdown or, for PRAs performed during preoperational phases,** as applicable, conditions through data and findings of investigation(s) of the plant (and its surroundings, as applicable to the hazard).
Table 4.3.11.1-6			
HS-E1 through HS-E4	-----------------------	No objection	---------------------------------
Section 4.3.11.2			
Section text	-----------------------	No objection	---------------------------------

Table A-12. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Hazard Screening Analysis

Index No.	Issue	Position	Resolution
Section 4.3.11.2.1			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.11.2.2			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.11.2.2.1			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.11.2.2.2			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.11.2.2.3			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.11.2.2.4			
Section text	-----------------------	No objection	--------------------------------

Table A-13. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for High Winds PRA

Index No.	Issue	Position	Resolution
Section 4.3.12			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.12.1			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.12.1-1			
HLR-WHA-A through HLR-WHA-G	-----------------------	No objection	---------------------------------
Table 4.3.12.1-2			
WHA-A1 through WHA-A4	-----------------------	No objection	---------------------------------
WHA-A5	The 150-mi distance in this requirement is too prescriptive, arbitrary, and based on the writers' judgement/experience. This doesn't conform to the revised ASME PRA, where the requirement was revised to be less arbitrary and more evidence based.	Clarification	… a. meet SCR-3 in Table 1.10-1 by showing that the site ~~is more than 150 miles (approximately 250 km)~~ **is sufficiently far away** from the nearest tropical cyclone-prone coast to screen out tropical cyclone (hurricane or typhoon) high wind hazards from the probabilistic wind hazard analysis;
WHA-A6 through WHA-A8	-----------------------	No objection	---------------------------------
Table 4.3.12.1-3			
WHA-B1 through WHA-B6	-----------------------	No objection	---------------------------------
Table 4.3.12.1-4			
WHA-C1 through WHA-C6	-----------------------	No objection	---------------------------------
Table 4.3.12.1-5			
WHA-D1 and WHA-D2	-----------------------	No objection	---------------------------------

Table A-13. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for High Winds PRA

Index No.	Issue	Position	Resolution
Table 4.3.12.1-6			
WHA-E1 through WHA-E5	-----------------------	No objection	---------------------------------
Table 4.3.12.1-7			
WHA-F1 through WHA-F4	-----------------------	No objection	---------------------------------
Table 4.3.12.1-8			
WHA-G1 through WHA-G4	-----------------------	No objection	---------------------------------
Table 4.3.12.2			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.12.2-1			
HLR-WFR-A through HLR-WFR-I	-----------------------	No objection	---------------------------------
Table 4.3.12.2-2			
WFR-A1	The performance of SSCs under high wind conditions that are not included in the plant response model, but that physically enclose or protect other SSCs included in the plant response model from high wind effects, needs to be considered. Without consideration of the performance of such SSCs, the plant response model may misrepresent how the plant would respond to failures of such SSCs due to high wind effects, which could therefore exclude	Clarification	INCLUDE in the scope of the Wind Fragility Analysis those SSCs and associated failure modes identified in the plant response analysis, **including those that may not be in the base plant model but that enclose or protect other SSCs from high wind effects.**

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Index No.	Issue	Position	Resolution
	important contributors to risk.		
WFR-A2 through WFR-A9	-----------------------	No objection	---------------------------------
Table 4.3.12.2-3			
WFR-B1 through WFR-B7	-----------------------	No objection	---------------------------------
Table 4.3.12.2-4			
WFR-C1 through WFR-C4	-----------------------	No objection	---------------------------------
Table 4.3.12.2-5			
WFR-D1 through WFR-D6	-----------------------	No objection	---------------------------------
Table 4.3.12.2-6			
WFR-E1 through WFR-E12	-----------------------	No objection	---------------------------------
Table 4.3.12.2-7			
WFR-F1 through WFR-F2	-----------------------	No objection	---------------------------------
Table 4.3.12.2-8			
WFR-G1 through WFR-G2	------------------	No objection	---------------------------------
Table 4.3.12.2-9			
WFR-H1 through WFR-H4	-----------------------	No objection	---------------------------------
Table 4.3.12.2-10			
WFR-I1	Documentation related to SSC failure mechanisms, the treatment of atmospheric pressure changes and missiles, structural interaction effects, wind-driven rain effects, and when	Clarification	***f*.) the identified SSC failure mechanisms and associated failure modes;** **g.) the treatment of wind pressure and atmospheric pressure change effects, wind -generated missile effects, structural interaction effects and wind -driven rain**

**Table A-13. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for High Winds PRA**

Index No.	**Issue**	**Position**	**Resolution**
	possible, the conduct of and observations from walkdowns is necessary to assess whether all important aspects of the high wind PRA are addressed. Without these aspects, the documentation would be incomplete and, therefore, insufficient to determine the acceptability of the high wind PRA.		**effects, if relevant to the plant;** **h.) walkdown observations following construction completion**
WFR-I2 and WFR-I3	------------------------	No objection	----------------------------------
Section 4.3.12.3			
Section text	------------------------	No objection	----------------------------------
Table 4.3.12.3-1			
HLR-WPR-A through HLR-WPR-F	------------------------	No objection	----------------------------------
Table 4.3.12.3-2			
WPR-A1 through WPR-A4	------------------------	No objection	----------------------------------
Table 4.3.12.3-3			
WPR-B1 through WPR-B9	------------------------	No objection	----------------------------------
Table 4.3.12.3-4			
WPR-C1 through WPR-C5	------------------------	No objection	--------------------------------
Table 4.3.12.3-5			
WPR-D1 through WPR-D6	------------------------	No objection	---------------------------------
WPR-D7	All operator actions relevant to the high wind plant response analysis should satisfy	Clarification	For treatment of additional~~,~~ ~~exclusive~~ **operator** recovery actions relevant to the Wind Plant Response Analysis,

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Index No.	Issue	Position	Resolution
	the appropriate HRA SRs. Satisfying the related HRA requirements only for certain operator recovery actions relevant to the Wind Plant Response analysis may exclude important contributors to risk.		SATISFY SRs of HLR HR-H, except where the requirements are not applicable.
WPR-D8 through WPR-D10	-----------------------	No objection	--------------------------------
WPR-D11	Failing to perform a feasibility assessment may result in some HEPs being assigned a value less than 1.0 when the assigned value should be 1.0. If an HFE that is not feasible is assigned an HEP value less than 1.0, but a feasibility assessment would otherwise demonstrate that the HFE is not feasible, the PRA model may underestimate the risk from that contributor.	Qualification	<u>CC-I</u> …When addressing **feasibility,** influencing factors, and the timing considerations in Requirements HR-G1, HR-G4, HR-G6, and HR-G8, INCLUDE the effect of high wind hazard on the control room and ex-control room human actions. <u>CC-II</u> …When addressing **feasibility,** influencing factors, and the timing considerations in Requirements HR-G1, HR-G4, HR-G6, and HR-G8…
Table 4.3.12.3-6			
WPR-E1 through WPR-E7	-----------------------	No objection	---------------------------------
Table 4.3.12.3-7			
WPR-F1 through WPR-F3	-----------------------	No objection	---------------------------------
Section 4.3.12.4			
Section 4.3.12.4.1			
Section text	-----------------------	No objection	---------------------------------

Table A-13. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for High Winds PRA

Index No.	Issue	Position	Resolution
Section 4.3.12.4.2			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.12.4.3			
Section 4.3.12.4.3.1			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.12.4.3.2			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.12.4.3.3			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.12.4.3.4			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.12.4.3.5			
Section text	-----------------------	No objection	--------------------------------

Table A-14. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for External Flooding PRA

Index No.	Issue	Position	Resolution
Section 4.3.13			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.13.1			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.13.1-1			
HLR-XFHA-A through HLR-XFHA-G	-----------------------	No objection	---------------------------------
Table 4.3.13.1-2			
XFHA-A1 through XFHA-A7	-----------------------	No objection	---------------------------------
XFHA-A8	A walkdown should be performed for post-operational plants as part of an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation that involves something less than a walkdown for post-operational plant may result in a mischaracterization of as well as missing information from the plant's as-built design in the PRA.	Clarification	CONFIRM that the external flood hazard screening represents either the as-built, as-operated or as-designed, as-intended-to-operate configuration of the plant, including relevant deficiencies (if applicable), and by performing **walkdowns or, for PRAs performed during the preoperational phase,** investigations.
XFHA-A9	-----------------------	No objection	---------------------------------
Table 4.3.13.1-3			
XFHA-B1 through XFHA-B4	-----------------------	No objection	---------------------------------
Table 4.3.13.1-4			

**Table A-14. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for External Flooding PRA**

Index No.	Issue	Position	Resolution
XFHA-C1 through XFHA-C11	-----------------------	No objection	---------------------------------
Table 4.3.13.1-5			
XFHA-D1 through XFHA-D4	-----------------------	No objection	---------------------------------
Table 4.3.13.1-6			
XFHA-E1 through XFHA-E4	-----------------------	No objection	---------------------------------
Table 4.3.13.1-7			
XFHA-F1	A walkdown should be performed for post-operational plants as part of an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation that involves something less than a walkdown for post-operational plant may result in a mischaracterization the plant's as-built design in the PRA.	Clarification	COLLECT information via **walkdown(s) or, for PRAs performed during the preoperational phase,** investigation(s) (complemented, as needed, by hydrologic surveys) about either the as-built, as-operated or as-designed, as-intended-to-operate, as applicable, plant and site characteristics relevant to the hazard analysis such as site topography, features that may affect flow around the site, drainage features, and features that may impound water.
XFHA-F2 through XFHA-F4	-----------------------	No objection	---------------------------------
Table 4.3.13.1-8			
XFHA-G1 through XFHA-G4	-----------------------	No objection	---------------------------------
Table 4.3.13.2			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.13.2-1			

Table A-14. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for External Flooding PRA

Index No.	Issue	Position	Resolution
HLR-XFFR-A through HLR-XFFR-F	------------------------	No objection	----------------------------------
Table 4.3.13.2-2			
XFFR-A1 through XFFR-A5	------------------------	No objection	----------------------------------
Table 4.3.13.2-3			

Table A-14. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for External Flooding PRA

Index No.	Issue	Position	Resolution
XFFR-B1	A walkdown should be performed for post-operational plants as part of an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation that involves something less than a walkdown for post-operational plant may result in a mischaracterization the plant's as-built design in the PRA.	Clarification	COLLECT information via **walkdown(s) or, for PRAs performed during the preoperational phase,** investigation(s) about either the as-built, as-operated or as-designed, as-intended-to-operate, as applicable, plant and site characteristics relevant to the fragility evaluation, such as establishing or confirming the location and characteristics of flood protection features, penetrations/seals, and drainage features.
XFFR-B2	A walkdown should be performed for post-operational plants as part of an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation that involves something less than a walkdown for post-operational plant may result in a mischaracterization the plant's as-built design in the PRA.	Clarification	For operating reactors, ASSESS the condition (e.g., SSC degradation) and configuration of SSCs observed during the **walkdowns or, for PRAs performed during the preoperational phase,** investigation(s).
XFFR-B3 through XFFR-B5	------------------------	No objection	----------------------------------
Table 4.3.13.2-4			
XFFR-C1 and XFFR-C2	------------------------	No objection	----------------------------------

**Table A-14. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for External Flooding PRA**

Index No.	Issue	Position	Resolution
Table 4.3.13.2-5			
XFFR-D1 through XFFR-D4	-----------------------	No objection	---------------------------------
Table 4.3.13.2-6			
XFFR-E1 through XFFR-E2	-----------------------	No objection	---------------------------------
Table 4.3.13.2-7			
XFFR-F1 through XFFR-F3	-----------------------	No objection	---------------------------------
Section 4.3.13.3			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.13.3-1			
HLR-XFPR-A through HLR-XFPR-H	-----------------------	No objection	---------------------------------
Table 4.3.13.3-2			
XFPR-A1 through XFPR-A7	-----------------------	No objection	---------------------------------
Table 4.3.13.3-3			
XFPR-B1 through XFPR-B3	-----------------------	No objection	---------------------------------
Table 4.3.13.3-4			
XFPR-C1 through XFPR-C12	-----------------------	No objection	---------------------------------
Table 4.3.13.3-5			
XFPR-D1 through XFPR-D5	-----------------------	No objection	---------------------------------
Table 4.3.14.3-6			
XFPR-E1 through XFPR-E5	-----------------------	No objection	---------------------------------
XFPR-E6	Failing to perform a feasibility assessment may result in some	Qualification	<u>CC-I</u> When addressing **feasibility,**

Table A-14. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for External Flooding PRA

Index No.	Issue	Position	Resolution
	HEPs being assigned a value less than 1.0 when the assigned value should be 1.0. If an HFE that is not feasible is assigned an HEP value less than 1.0, but a feasibility assessment would otherwise demonstrate that the HFE is not feasible, the PRA model may underestimate the risk from that contributor.		influencing factors**,** and the timing considerations in Requirements **HR-G1,** HR-G4, HR-G6, and HR-G8, INCLUDE the effect of external flooding hazard on the control room and ex-control room human actions. CC-II When addressing **feasibility,** influencing factors**,** and the timing considerations in Requirements **HR-G1,** HR-G4, HR-G6, and HR-G8, INCLUDE the effect of ~~flood impacts~~ **external flooding hazard** on the control room and ex-control room human actions.
XFPR-E7 and XFPR-E8	------------------------	No objection	----------------------------------
Table 4.3.13.3-7			
XFPR-F1 through XFPR-F7	------------------------	No objection	----------------------------------
Table 4.3.13.3-8			
XFPR-G1	A walkdown should be performed for post-operational plants as part of an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation that involves something less than a walkdown for post-operational plant may result in a mischaracterization	Clarification	COLLECT information via **walkdown(s) or, for PRAs performed during pre-operational phase,** investigation(s) about either the as-built, as-operated or as-designed, as-intended-to-operate, as applicable, plant and site characteristics relevant to the plant response analysis including consideration of operations; flooding design/licensing basis; design, construction, and performance of flood protection features; hydrology and hydraulics; and manual actions.

Table A-14. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for External Flooding PRA

Index No.	Issue	Position	Resolution
	the plant's as-built design in the PRA.		
XFPR-G2	-----------------------	No objection	--------------------------------
XFPR-G3	A walkdown should be performed for post-operational plants as part of an investigation. However, a physical walkdown of a plant in the pre-operational phase of development may not always be possible. Performing an investigation that involves something less than a walkdown for post-operational plant may result in a mischaracterization the plant's as-built design in the PRA.	Clarification	EVALUATE the external floods plant response analysis for the as-built, as-operated or as-designed, as-intended-to-operate plant conditions via **walkdown(s) or, for PRAs performed during pre-operational phase,** investigation(s) to do the following:
Table 4.3.13.3-9			
XFPR-H1 through XFPR-H3	-----------------------	No objection	--------------------------------
Section 4.3.13.4			
Section 4.3.13.4.1			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.13.4.2			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.13.4.3			
Section 4.3.13.4.3.1			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.13.4.3.2			
Section 4.3.13.4.3.2.1			
Section text	-----------------------	No objection	--------------------------------

Table A-14. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for External Flooding PRA

Index No.	Issue	Position	Resolution
Section 4.3.13.4.3.2.2			
Section text	------------------------	No objection	---------------------------------
Section 4.3.13.4.3.2.3			
Section text	------------------------	No objection	---------------------------------
Section 4.3.13.4.3.2.3.1			
Section text	------------------------	No objection	---------------------------------
Section 4.3.13.4.3.3.3.2			
Section text	------------------------	No objection	---------------------------------
Section 4.3.13.4.3.3.3.3			
Section text	------------------------	No objection	---------------------------------

Table A-15. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Other Hazards PRA

Index No.	Issue	Position	Resolution
Section 4.3.14			
Section text	------------------------	No objection	----------------------------------
Section 4.3.14.1			
Section text	------------------------	No objection	----------------------------------
Table 4.3.14.1-1			
HLR-OHA-A through HLR-OHA-B	------------------------	No objection	----------------------------------
Table 4.3.14.1-2			
OHA-A1 through OHA-A10	------------------------	No objection	----------------------------------
Table 4.3.14.1-3			
OHA-B1 through OHA-B4	------------------------	No objection	----------------------------------
Table 4.3.14.2			
Section text	------------------------	No objection	----------------------------------
Table 4.3.14.2-1			
HLR-OFR-A through HLR-OFR-B	------------------------	No objection	----------------------------------
Table 4.3.14.2-2			
OFR-A1 through OFR-A7	------------------------	No objection	----------------------------------
Table 4.3.14.2-3			
OFR-B1 through OFR-B3	------------------------	No objection	----------------------------------
Section 4.3.14.3			
Section text	------------------------	No objection	----------------------------------
Table 4.3.14.3-1			
HLR-OPR-A through HLR-OPR-E	------------------------	No objection	----------------------------------

**Table A-15. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Other Hazards PRA**

Index No.	Issue	Position	Resolution
Table 4.3.14.3-2			
OPR-A1 through OPR-A4	------------------------	No objection	----------------------------------
Table 4.3.14.3-3			
OPR-B1 through OPR-B12	------------------------	No objection	----------------------------------
Table 4.3.14.3-4			
OPR-C1 through OPR-C5	------------------------	No objection	----------------------------------
OPR-C6	Failing to perform a feasibility assessment may result in some HEPs being assigned a value less than 1.0 when the assigned value should be 1.0. If an HFE that is not feasible is assigned an HEP value less than 1.0, but a feasibility assessment would otherwise demonstrate that the HFE is not feasible, the PRA model may underestimate the risk from that contributor.	Qualification	CC-I: **ASSESS the feasibility of the HFE using the criteria in HR-H2. If the HFE is not feasible, ASSIGN an HEP of 1.0 or DO NOT CREDIT the HFE in the PRA. For HFEs determined to be feasible,** USE screening values in accordance with Requirement OPR-C5 for the HEPs for HFEs included in the hazard PRA model. CC-II: … Attention is to be given to how the hazard situation alters previous assessments in non-hazard analyses as to the **feasibility,** influencing factors**,** and the timing considerations in Requirements **HR-G1,** HR-G4, HR-G6, and HR-G8 except when they are not applicable.
OPR-C7 through OPR-C8	------------------------	No objection	----------------------------------
Table 4.3.14.3-5			

Table A-15. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Other Hazards PRA

Index No.	Issue	Position	Resolution
OPR-D1 through OPR-D9	------------------------	No objection	----------------------------------
Table 4.3.14.3-6			
OPR-E1 through OPR-E5	------------------------	No objection	----------------------------------
Section 4.3.14.4			
Section 4.3.14.4.1			
Section text	------------------------	No objection	----------------------------------
Section 4.3.14.4.2			
Section text	------------------------	No objection	----------------------------------
Section 4.3.14.4.3			
Section 4.3.14.4.3.1			
Section text	------------------------	No objection	----------------------------------
Section 4.3.14.4.3.2			
Section text	------------------------	No objection	----------------------------------
Section 4.3.14.4.3.3			
Section text	------------------------	No objection	----------------------------------
Section 4.3.13.4.3.4			
Section text	------------------------	No objection	----------------------------------
Section 4.3.13.4.3.5			
Section text	------------------------	No objection	----------------------------------

Table A-16. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Event Sequence Quantification

Index No.	Issue	Position	Resolution
Section 4.3.15			
Section text	------------------------	No objection	----------------------------------
Section 4.3.15.1			
Section text	------------------------	No objection	----------------------------------
Table 4.3.15.1-1			
HLR-ESQ-A through HLR-ESQ-F	------------------------	No objection	----------------------------------
Table 4.3.15.1-2			
ESQ-A1 through ESQ-A9	------------------------	No objection	----------------------------------
Table 4.3.15.1-3			
ESQ-B1 through ESQ-B10	------------------------	No objection	----------------------------------
Table 4.3.15.1-4			
ESQ-C1 through ESQ-C17	------------------------	No objection	----------------------------------
Table 4.3.15.1-5			
ESQ-D1 through ESQ-D8	------------------------	No objection	----------------------------------
Table 4.3.15.1-6			
ESQ-E1 through ESQ-E2	------------------------	No objection	----------------------------------
Table 4.3.15.1-7			
ESQ-F1 through ESQ-F5	------------------------	No objection	----------------------------------
Section 4.3.15.2			
Section text	------------------------	No objection	----------------------------------
Section 4.3.15.2.1			
Section text	------------------------	No objection	----------------------------------
Section 4.3.15.2.2			

Table A-16. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Event Sequence Quantification

Index No.	Issue	Position	Resolution
Section text	------------------------	No objection	----------------------------------
Section 4.3.15.2.3			
Section text	------------------------	No objection	----------------------------------

Table A-17. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Mechanistic Source Term Analysis

Index No.	Issue	Position	Resolution
Section 4.3.16			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.16.1			
Section text	-----------------------	No objection	--------------------------------
Table 4.3.16.1-1			
HLR-MS-A through HLR-MS-E	-----------------------	No objection	--------------------------------
Table 4.3.16.1-2			
MS-A1 through MS-A5	-----------------------	No objection	--------------------------------
Table 4.3.16.1-3			
MS-B1 through MS-B7	-----------------------	No objection	--------------------------------
Table 4.3.16.1-4			
MS-C1 through MS-C7	-----------------------	No objection	--------------------------------
Table 4.3.16.1-5			
MS-D1 through MS-D4	-----------------------	No objection	--------------------------------
Table 4.3.16.1-6			
MS-E1 through MS-E4	-----------------------	No objection	--------------------------------
Section 4.3.16.2			
Section 4.3.16.2.1			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.16.2.2			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.16.2.3			
Section text	-----------------------	No objection	--------------------------------

Table A-18. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Radiological Consequence Analysis

Index No.	**Issue**	**Position**	**Resolution**
Section 4.3.17			
Section text	-----------------------	No objection	---------------------------------
Section 4.3.17.1			
Section text	-----------------------	No objection	---------------------------------
Table 4.3.17.1-1			
HLR-RCRE-A through HLR-RCRE-C	-----------------------	No objection	---------------------------------
Table 4.3.17.1-2			
RCRE-A1	-----------------------	No objection	---------------------------------
RCRE-A2	There is no requirement to identify the radiological inventory, only the release fraction. This is an issue for PRA quality because the consequences are dependent on the quantity of radionuclides released, which depends upon the inventory as well as the release fraction. The inventory at the time of accident initiation can change over the operating cycle because of, for example, the ingrowth of important long-lived fission products such radiocesium or the decay in important short lived fission products such as radioiodine during and after coastdown.	Qualification	… At a minimum, INCLUDE the following characteristics for each release category, if applicable: (a) the number of plumes; (b) **the quantity of radionuclides released by species in each time phase of release.** ~~the release fraction of each radionuclide group~~; …
RCRE-A3	-----------------------	No objection	---------------------------------
Table 4.3.17.1-3			
RCRE-B1 and RCRE-B2	-----------------------	No objection	---------------------------------

Table A-18. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Radiological Consequence Analysis

Index No.	**Issue**	**Position**	**Resolution**
Table 4.3.17.1-4			
RCRE-C1	------------------------	No objection	----------------------------------
Section 4.3.17.2			
Section text	------------------------	No objection	----------------------------------
Table 4.3.17.2-1			
HLR-RCPA-A through HLR-RCPA-C	------------------------	No objection	----------------------------------
Table 4.3.17.2-2			
RCPA-A1 through RCPA-A14	------------------------	No objection	----------------------------------
Table 4.3.17.2-3			
RCPA-B1 through RCPA-B8	------------------------	No objection	----------------------------------
Table 4.3.17.2-4			
RCPA-C1 through RCPA-C3	------------------------	No objection	----------------------------------
Section 4.3.17.3			
Section text	------------------------	No objection	----------------------------------
Table 4.3.17.3-1			
HLR-RCME-A and HLR-RCME-B	------------------------	No objection	----------------------------------
Table 4.3.17.3-2			
RCME-A1 through RCME-A11	-------------------------	No objection	-----------------------------------
Table 4.3.17.3-3			
RCME-B1 through RCME-B3	------------------------	No objection	----------------------------------
Section 4.3.17.4			
Section text	------------------------	No objection	----------------------------------
Table 4.3.17.4-1			

Table A-18. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Radiological Consequence Analysis

Index No.	Issue	Position	Resolution
HLR-RCAD-A through HLR-RCAD-F	------------------------	No objection	----------------------------------
Table 4.3.17.4-2			
RCAD-A1 through RCAD-A8	------------------------	No objection	----------------------------------
Table 4.3.17.4-3			
RCAD-B1 and RCAD-B2	------------------------	No objection	----------------------------------
Table 4.3.17.4-4			
RCAD-C1 through RCAD-C6	------------------------	No objection	----------------------------------
Table 4.3.17.4-5			
RCAD-D1 through RCAD-D4	------------------------	No objection	----------------------------------
Table 4.3.17.4-6			
RCAD-E1 through RCAD-E7	------------------------	No objection	----------------------------------
Table 4.3.17.4-7			
RCAD-F1 through RCAD-F3	------------------------	No objection	----------------------------------
Section 4.3.17.5			
Section text	------------------------	No objection	----------------------------------
Table 4.3.17.5-1			
HLR-RCDO-A	Skin absorption could be a significant pathway for isotopes that may be important for some designs (e.g., tritium). This is an issue for PRA quality because omission of skin absorption could underestimate doses in such cases.	Qualification	The analysis shall include applicable exposure pathways including cloudshine, groundshine, skin deposition, **skin absorption,** inhalation and ingestion, and the effect of mitigation actions on received dose.

**Table A-18. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Radiological Consequence Analysis**

Index No.	Issue	Position	Resolution
HLR-RCDO-B and HLR-RCDO-C	------------------------	No objection	---------------------------------
Table 4.3.17.5-2			
RCDO-A1	The list of exposure pathways does not include skin absorption. This could be a significant pathway for isotopes that may be important for some designs (e.g., tritium). This is an issue for PRA quality because omission of skin absorption could underestimate doses in such cases.	Qualification	… JUSTIFY excluding any of the following pathways: (a) cloudshine; (b) groundshine; (c) skin deposition; (d) inhalation; (e) ingestion**;** **(f) skin absorption**
RCDO-A2 through RCDO-A5	------------------------	No objection	---------------------------------
RCDO-A6	This supporting requirement conflicts with the requirements of RCDO-A1, which requires the analyst to justify exclusion of the skin absorption and deposition pathways. This is an issue for PRA quality because the exclusion of these pathways without justification may result in underestimation of exposures if the excluded pathways were risk-significant	Qualification	CC-I ~~DO NOT INCLUDE skin deposition pathway in the model.~~ CC-II ~~INCLUDE skin deposition and beta exposure to the skin from the plume in the model~~ CC-I and CC-II **MODEL skin absorption and deposition pathways.**
RCDO-A7	------------------------	No objection	----------------------------------
RCDO-A8	This supporting requirement conflicts with the requirements of RCDO-A1, which requires the analyst to justify exclusion of ingestion pathways. This	Qualification	CC-I ~~DO NOT INCLUDE ingestion pathways in the model.~~ CC-II ~~USE generic intake quantities~~

Table A-18. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Radiological Consequence Analysis

Index No.	**Issue**	**Position**	**Resolution**
	is an issue for PRA quality because the exclusion of this pathway without justification may result in underestimation of exposure if the excluded pathway is risk-significant		~~of foodstuffs and water.~~ CC-I and CC-II **USE typical or average intake quantities of foodstuffs and water reflecting national or regional dietary patterns.** **USE national or regional agricultural productivity input parameters.**
RCDO-A9 and RCDO-A10	------------------------	No objection	----------------------------------
Table 4.3.17.5-3			
RCDO-B1 and RCDO-B2	------------------------	No objection	----------------------------------
Table 4.3.17.5-4			
RCDO-C1 and RCDO-C2	------------------------	No objection	----------------------------------
Section 4.3.17.6			
Section text	------------------------	No objection	----------------------------------
Table 4.3.17.6-1			
HLR-RCHE-A through HLR-RCHE-C	------------------------	No objection	----------------------------------
Table 4.3.17.6-2			
RCHE-A1 through RCHE-A7	------------------------	No objection	----------------------------------
Table 4.3.17.6-3			
RCHE-B1 through RCHE-B3	------------------------	No objection	----------------------------------
Table 4.3.17.6-4			
RCHE-C1 through RCHE-C3	------------------------	No objection	----------------------------------
Section 4.3.17.7			

Table A-18. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Radiological Consequence Analysis

Index No.	Issue	Position	Resolution
Section text	-----------------------	No objection	--------------------------------
Table 4.3.17.7-1			
HLR-RCEC-A through HLR-RCEC-C	-----------------------	No objection	--------------------------------
Table 4.3.17.7-2			
RCEC-A1 and RCEC-A2	-----------------------	No objection	--------------------------------
Table 4.3.17.7-3			
RCEC-B1 through RCEC-B7	-----------------------	No objection	--------------------------------
Table 4.3.17.7-4			
RCEC-C1 through RCEC-C3	-----------------------	No objection	--------------------------------
Section 4.3.17.8			
Section text	-----------------------	No objection	--------------------------------
Table 4.3.17.8-1			
HLR-RCQ-A through HLR-RCQ-D	-----------------------	No objection	--------------------------------
Table 4.3.17.8-2			
RCQ-A1 through RCQ-A3	-----------------------	No objection	--------------------------------
Table 4.3.17.8-3			
RCQ-B1 through RCQ-B3	-----------------------	No objection	--------------------------------
Table 4.3.17.8-4			
RCQ-C1 and RCQ-C2	-----------------------	No objection	--------------------------------
Table 4.3.17.8-5			
RCQ-D1 through RCQ-D3	-----------------------	No objection	--------------------------------

**Table A-18. Staff Position on ASME/ANS RA-S-1.4-2021,
Technical Requirements for Radiological Consequence Analysis**

Index No.	Issue	Position	Resolution
Section 4.3.17.9			
Section 4.3.17.9.1			
Section text	------------------------	No objection	----------------------------------
Section 4.3.17.9.2			
Section text	------------------------	No objection	----------------------------------
Section 4.3.17.9.3			
Section text	------------------------	No objection	----------------------------------

Table A-19. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Risk Integration

Index No.	Issue	Position	Resolution
Section 4.3.18			
Section text	------------------------	No objection	----------------------------------
Section 4.3.18.1			
Section text	------------------------	No objection	----------------------------------
Table 4.3.18.1-1			
HLR-RI-A through HLR-RI-D	------------------------	No objection	----------------------------------
Table 4.3.18.1-2			
RI-A1 through RI-A3	------------------------	No objection	----------------------------------
RI-A4	Using generic reporting requirements in place of application-specific reporting requirements may result in information important to the decision-making process not being provided. Such reporting requirements should be provided by the appropriate regulatory authority on an application-specific basis.	Qualification	USE ~~a~~ **application-specific** minimum reporting **guidance** ~~frequency of 10-7 per plant-year~~ for all modeled event sequence families or JUSTIFY an alternative.
RI-A5	Using generic reporting requirements in place of application-specific reporting requirements may result in information important to the decision-making process not being provided and consequences may not be expressed in terms of dose units. Such reporting requirements should be provided by the appropriate regulatory authority on	Qualification	USE ~~a~~ **application-specific** minimum reporting **guidance** ~~consequence of 10% of the consequences due to background radiation dose~~ for consequence metrics associated with all modeled event sequence families or JUSTIFY an alternative.

Table A-19. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Risk Integration

Index No.	Issue	Position	Resolution
	an application-specific basis.		
Table 4.3.18.1-3			
RI-B1 through RI-B3	-----------------------	No objection	--------------------------------
RI-B4	Use of the term "unscreened" may imply that an item was previously screened out and subsequently screened back into the PRA. However, this term is interpreted as meaning sources of radioactive material that were included in the PRA for consideration and evaluation (i.e., screened in).	Clarification	INCLUDE the risk contributions from modeled event sequence families involving releases from ~~multiple reactors and unscreened~~ **all** sources of radioactive material **modeled in** ~~included in the scope of~~ the PRA.
RI-B5 through RI-B7	-----------------------	No objection	--------------------------------
Table 4.3.18.1-4			
RI-C1 through RI-C4	-----------------------	No objection	--------------------------------
Table 4.3.18.1-5			
RI-D1 through RI-D2	-----------------------	No objection	--------------------------------
Section 4.3.18.2			
Section 4.3.18.2.1			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.18.2.2			
Section text	-----------------------	No objection	--------------------------------
Section 4.3.18.2.3			
Section text	-----------------------	No objection	--------------------------------

Table A-20. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for PRA Configuration Control

Index No.	Issue	Position	Resolution
Section 5			
Section 5.1			
Section text	-----------------------	No objection	---------------------------------
Section 5.2			
Section text	-----------------------	No objection	---------------------------------
Table 5.8-1			
HLR-CC-A through HLR-CC-E	-----------------------	No objection	---------------------------------
Table 5.8-2			
CC-A1 through CC-A6	-----------------------	No objection	---------------------------------
Table 5.8-3			
CC-B1 through CC-B5	-----------------------	No objection	---------------------------------
Table 5.8-4			
CC-C1 and CC-C2	-----------------------	No objection	---------------------------------
Table 5.8-5			
CC-D1	-----------------------	No objection	---------------------------------
Table 5.8-6			
CC-E1 and CC-E2	-----------------------	No objection	---------------------------------

Table A-21. Staff Position on ASME/ANS RA-S-1.4-2021, Technical Requirements for Peer Review

Index No.	Issue	Position	Resolution
Section 6			
Section 6.1			
Section text	------------------------	No objection	----------------------------------
Section 6.2			
Section text	------------------------	No objection	----------------------------------
Section 6.3			
Section text	------------------------	No objection	----------------------------------
Section 6.4			
Section text	------------------------	No objection	----------------------------------
Section 6.5			
Section text	------------------------	No objection	----------------------------------
Section 6.6			
Section text	------------------------	No objection	----------------------------------

Table A-22. Staff Position on ASME/ANS RA-S-1.4-2021, Newly Developed Methods

Index No.	Issue	Position	Resolution
Section 7			
Section 7.1			
Section text	------------------------	No objection	------------------------------
Section 7.2			
Section text	------------------------	No objection	------------------------------
Table 7.2-1			
HLR-NM-A through HLR-NM-F	------------------------	No objection	------------------------------
Table 7.2-2			
NM-A1 through NM-A3	-----------------------	No objection	------------------------------
Table 7.2-3			
NM-B1 through NM-B4	-----------------------	No objection	---------------------------------
Table 7.2-4			
NM-C1 through NM-C6	-----------------------	No objection	---------------------------------
Table 7.2-5			
NM-D1 through NM-D3	-----------------------	No objection	---------------------------------
Table 7.2-6			
NM-E1 through NM-E3	------------------------	No objection	---------------------------------
Table 7.2-7			
NM-F1	Any manipulation of data performed to support an NDM should be documented along with the sources of data and the collection process. Without documenting how data may have been manipulated, the impacts of that manipulation	Clarification	*(e)* the sources of data, and the collection process **and data manipulation performed** in support of the Newly Developed Method~~s~~;

Table A-22. Staff Position on ASME/ANS RA-S-1.4-2021, Newly Developed Methods

Index No.	Issue	Position	Resolution
	may not be understood and, depending on how the data is manipulated, the use of the NDM in the PRA could potentially underestimate risk.		
NM-F2	-------------------------	No objection	----------------------------------

REFERENCES[1]

1. American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants," ASME, New York, NY, ANS, La Grange Park, IL, February 2021.

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

1 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

HAZARDS FOR CONSIDERATION IN A PROBABILISTIC RISK ASSESSMENT

A key feature of a 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 ensure that significant contributors to plant risk are not inadvertently excluded. 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 or external, 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 quantitatively using a PRA. However, there are 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.1.3.11 of this regulatory guide (RG) provides additional guidance on screening and conservative analyses used to screen hazards from a detailed PRA. Conversely, some such internal and external hazards may produce impacts on 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.1.3.14 provides additional guidance on modeling 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 B-1 and B-2. Table B-1 provides a list of hazards consistent with those in Table HS-2 of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) advanced nonlight-water reactor PRA standard, ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants" (Ref. 1). Table B-1 also provides a general description of direct and indirect impacts of each hazard within a hazard group that should be considered during the development of a PRA. The genesis of this list of hazards 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 1983 (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 B-2 lists hazard causes and potential conditions to consider during the process of determining what risk a given hazard poses to a facility. The taxonomy of these hazard groups and the hazards within those groups are relevant to applications only.

Table B-1. List of Hazards

Hazard	Hazard Group	Direct or Secondary Impact of Hazard
Animals	*Animals*	Land-based or airborne animals that cause damage to plant equipment, such as loss of offsite power, or that result in other hazards (such as transportation accidents).
Biological Events	*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.
External Fire	*Wildfire*	Direct (e.g., thermal effects) or indirect effects (e.g., generation of combustion products) of a fire in an area of combustible vegetation (e.g., trees, grass) outside the plant boundary defined by the internal fire PRA.
	Non-safety-related Building Fire	Direct (e.g., thermal effects) or indirect effects (e.g., generation of combustion products, propagation to safety-related structures, systems, and components (SSCs) of a fire in a non-safety-related building.
External Flooding	*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. This hazard may be analyzed when it occurs concurrent with other hazards such as a storm surge or straight wind to produce flooding effects.
	Hurricane Flooding	Flooding that results from a hurricane (tropical cyclone). For example, storm surge, flooding due to rivers and streams, flooding due to dam failure, flooding due to intense rainfall, and flooding due to a wind-caused seiche, as induced by a hurricane.
	Local Intense Precipitation	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.
	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.

Hazard	Hazard Group	Direct or Secondary Impact of Hazard
	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, river and stream overflow and waves.
	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 slide or landslide. Secondary hazards resulting from a tsunami include, but are not necessarily limited to, river and stream overflow.
	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. This hazard is typically analyzed in the context of other concurrent events like a hurricane.
Extraterrestrial Events	*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.
Extreme Temperature	*High Summer Temperature*	Effects on SSC operation due to abnormally high ambient temperatures resulting from weather phenomena or other causes. Secondary hazards resulting from high ambient temperatures include, but are not necessarily limited to, low lake or river water levels.
	Ice	Reduced flow or blockage of water systems due to the accumulation of ice on or in (i.e., frazil ice) a body of water (e.g., lakes, rivers, ocean) or the water system itself. This hazard is also analyzed for the effects of static loading of SSCs due to ice accumulation.
	Low Winter Temperature	Effects on SSC operation due to abnormally low ambient temperatures resulting from weather phenomena or other causes. Secondary hazards resulting from low ambient temperatures include, but are not necessarily limited to, frost, ice, and snow.

Hazard	Hazard Group	Direct or Secondary Impact of Hazard
Ground Shifts	*Coastal Erosion*	Natural removal of earth 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 may impact the structural integrity of SSCs.
	Landslide	Rapid flow of a large mass of earth or other debris (e.g., mud) 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.
	Sinkholes	Ground movement effects on SSC structural integrity due to karst (i.e., topography formed by the dissolution of soluble rocks).
	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.
Heat Sink Effects	*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.
	Low Lake or River Water Level	A decrease in the water level of the lake or river used for power generation.
	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).
Heavy Load Drop Hazards	*Heavy Load Drop*	An uncontrolled, unplanned lowering of a heavy load onto an SSC. This hazard is analyzed with respect to direct and indirect effects on SSCs.
High Wind	*Hurricane Winds*	Dynamic loading on SSCs from wind or missiles due to a hurricane.
	Straight Winds	Dynamic loading on SSCs from wind or missiles due to a strong wind that is not associated with either tornadoes or hurricanes (e.g., derecho).
	Tornado	Dynamic loading on SSCs from wind or missiles due to a tornado.
	Sandstorm	Persistent heavy winds transporting sand or dust that infiltrates SSCs at or in the plant's analyzed area causing functional failure.
	Hail	A shower of ice or hard snow that could result in transportation accidents or directly cause dynamic loading or freezing conditions as a result of ice coverage.

Hazard	Hazard Group	Direct or Secondary Impact of Hazard
Industrial Accidents	*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.
	Onsite Excavation Work	The unintended effects of onsite excavation work that may impact the structural integrity of SSCs.
	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.
	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.
	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.
Lightning	*Lightning*	Effects on SSCs due to a sudden electrical discharge from a cloud to the ground or Earth-bound object.
Seismic	*Natural Tectonic Earthquakes*	Sudden natural 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, dam failure, industrial accidents, landslide, seiche, tsunami, and vehicle accidents.
	Human-Induced Earthquakes	Sudden human-induced 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, dam failure, industrial accidents, landslide, seiche, tsunami, and vehicle accidents.
Snow	*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.

Hazard	Hazard Group	Direct or Secondary Impact of Hazard
	Snow Cover	The accumulation of snow could result in transportation accidents or directly cause dynamic loading or freezing conditions as a result of snow cover.
Site-Generated Missiles	*Turbine-Generated Missiles*	Damage to SSCs from a missile generated internal or external to the plant PRA boundary from rotating turbines. Damage may result from a falling missile or a missile ejected directly toward SSCs (i.e., low-trajectory missiles).
	Missiles Generated from Other Sources	Damage to SSCs due to a missile generated from sources other than a turbine, such as high-pressure gas cylinders.
Transportation Accidents	*Aircraft Impact*	An aircraft (either a portion of or the entire aircraft) that collides either directly or indirectly (i.e., skidding impact) with one or more 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.
	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.
	Frost	A thin layer of ice crystals that forms on the ground or on the surface of an earth-bound object when the temperature of the ground or surface of the object falls below freezing. This hazard could result in a transportation accident.
	Railcar Impact	Effects of an onsite railcar impact with one or more SSCs.
	Ship Impacts	Effects of a waterborne vessel impact with a water intake or outlet SSC.
	Vehicle Impacts	Effects of an onsite vehicle impact with one or more SSCs.
	Transportation Vehicle Explosion	Offsite accidents involving transportation resulting in a release of hazardous materials or combustion products, an explosion, or the generation of missiles causing functional failure of SSCs or preventing operator actions. 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.

Hazard	Hazard Group	Direct or Secondary Impact of Hazard
Volcanic Activity	*Volcanic Activity*	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.

Table B-2. List of Hazard Causes and Conditions

<table>
<tr><td>Combustion/fire—resulting in burning, release of hot/toxic gases, release of combustion products, or heat causing functional failure of SSCs, or failure of the operator to perform, or both.</td></tr>
<tr><td>Debris effects—resulting in clogging of a liquid flow path or adversely affecting equipment performance.</td></tr>
<tr><td>Dynamic forces (from dynamic or static loading)—resulting in structural damage to SSCs causing functional failure of SSCs, or impeded operator ability to perform actions, or both.</td></tr>
<tr><td>Explosions—resulting in dynamic forces, fire, missiles, or gas releases.</td></tr>
<tr><td>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)</td></tr>
<tr><td>High-energy arcs</td></tr>
<tr><td>Missiles—projectiles damaging structures and equipment</td></tr>
<tr><td>Physical obstruction—movement of structures or equipment that reduces accessibility</td></tr>
<tr><td>Reduced air quality—combustion products or other airborne particulates affecting equipment or operator performance</td></tr>
<tr><td>Reduced availability of cooling water</td></tr>
<tr><td>Structural failure, including the following:
collapse
functional failure (e.g., break in containment, settlement of a structure)
loss of structural integrity</td></tr>
<tr><td>Thermal effects, including the following:
heat transfer (radiative, conductive, or convective; advection (i.e., bulk transport of a fluid))
steam</td></tr>
<tr><td>Water effects—water infiltration, submergence or spray causing corrosion, loss of electrical integrity (e.g., electrical short), clogging, inaccessibility, structural failure</td></tr>
</table>

REFERENCES[1]

1. American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants," ASME, New York, NY, ANS, La Grange Park, IL, February 2021.

2. U.S. Nuclear Regulatory Commission (NRC), NUREG/CR-2300, "PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants," Washington, DC, January 1983. (Agencywide Documents Access and Management System (ADAMS) Accession Package No. ML063560438)

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

[1] 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.