U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE RG 1.247

(For Trial Use)

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Technical Lead: Michelle Gonzalez

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

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes one trial approach the U.S. Nuclear Regulatory Commission (NRC) staff has developed for determining whether a design-specific or plant-specific probabilistic risk assessment (PRA) uioniosed to support an application is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision-making for advanced non-light-water reactors (NLWRs). In this RG, the term application includes pre-application activities, 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 NRC reviewers, allowing 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.

This RG 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 and clarifications, a national consensus PRA standard provided by standards development organizations, and guidance provided by nuclear industry organizations.

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. 3). Specifically:

- standard design certification (DC): 10 CFR Part 52, Subpart B
- combined license (COL): 10 CFR Part 52, Subpart C
- standard design approval (SDA): 10 CFR Part 52, Subpart E
- manufacturing license (ML): 10 CFR Part 52, Subpart F

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

This RG also applies to applications for NLWR licensing under Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 2) if performed. Specifically:

- 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.

- 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities" 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 combined license 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 one year prior to the scheduled date for initial loading of fuel.
    - 10 CFR 50.71(h)(2), requires that (1) each holder of a combined license 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 one year prior to each required upgrade, and (3) the PRA must be upgraded every four 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 combined license 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, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The regulations in this part govern the issuance of early site permits, standard design certifications, combined licenses, standard design approvals, and manufacturing licenses for nuclear power facilities pursuant to the Atomic Energy Act of 1954, as amended and Title II of the Energy Reorganization Act of 1974.
    - 10 CFR 52.47(a)(27), requires that applicants for a standard design certification under Subpart B of 10 CFR Part 52 shall provide a description of the design-specific probabilistic risk assessment (PRA) and its results.
    - 10 CFR 52.79(a)(46), requires that applicants for a combined license under Subpart C of 10 CFR Part 52 shall provide a description of the plant-specific probabilistic risk assessment (PRA) and its results.
    - 10 CFR 52.79(c)(1), requires that applicants for a combined license under Subpart C of 10 CFR Part 52 who reference a standard design approval under Subpart E of 10 CFR Part 52 shall use and update the PRA information for the standard design approval to account for site-specific design information and any design changes or departures.

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

- 10 CFR 52.79(e)(1), requires that applicants for a combined license under Subpart C of 10 CFR Part 52 who 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 a standard design approval under Subpart E of 10 CFR Part 52 shall include a description of the design-specific probabilistic risk assessment and its results.

- 10 CFR 52.157(a)(31) requires that applicants for a manufacturing license under Subpart F of 10 CFR Part 52 shall include a description of the design-specific probabilistic risk assessment and its results.

Related Guidance

- NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 8), 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 early site permit (ESP), DC, COL, standard SDA, and ML applications under 10 CFR Part 52 (including requests for amendments).

  - NUREG-0800, Section 19.1, titled "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Ref. 9) 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 regulatory guides (RGs) and the corresponding chapters of NUREG-0800 also applies to these types of licensing basis changes.

  - NUREG-0800, Section 19.0, titled "Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors" (Ref. 10), pertains to the NRC staff review of the design-specific PRA for a DC and plant-specific PRA for a COL application, respectively.

- NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," issued March 2017 (Ref. 11), provides guidance on how to treat uncertainties associated with PRA in risk-informed decisionmaking. This guidance is intended to foster an understanding of the uncertainties associated with PRA and their impact on the results of PRA.

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Ref. 12), provides guidance on an acceptable approach for developing risk-informed applications for a licensing basis change that considers engineering issues and applies risk insights.

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- RG 1.200 , "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," (Ref. 13) provides an acceptable approach for determining whether a base probabilistic risk assessment (PRA), in total or in the portions that are used to support an application, is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decisionmaking for light-water reactors (LWRs). When used in support of an application, this RG will obviate the need for an in-depth review of the base PRA by NRC reviewers, allowing them to focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application. Consequently, RG 1.200 provides for a more focused and consistent review process.

- RG 1.206, "Applications for Nuclear Power Plants" (Ref. 14) provides guidance on the format and content of applications for nuclear power plants submitted to the NRC under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," 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 early site permits, standard design certifications, and combined licenses 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.15), 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. This guidance may be used by NLWR applicants applying for permits, licenses, certifications, and approvals under 10 CFR Parts 50 and Part 52. RG 1.233 endorses the licensing modernization project (LMP) guidance provided in Nuclear Energy Institute (NEI) 18-04.

- 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. 16), provides guidance on how to adapt the requirements[1] provided in the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) PRA standard ASME/ANS RA-Sa-2009, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Ref. 17), which was developed for currently operating reactors, to PRAs that support advanced LWR standard design certification and combined license applications.

Purpose of Regulatory Guides

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

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

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required for the issuance or continuance of a permit or license by the Commission. For more information on the intended use of this draft regulatory guide, please see "D. Implementation" below.

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), approval numbers 3150-0011 and 3150-0151, 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 an approach acceptable to the NRC staff 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, 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. 21) and the NRC developed the endorsement thereof in this RG. As part of the NRC's efforts to endorse ASME/ANS RA-S-1.4-2021, the staff considered several endorsement vehicle options that could be used to promulgate the staff position on the standard. After considering a variety of options, the staff determined that a regulatory guide would be most appropriate for endorsing ASME/ANS RA-S-1.4-2021, which is based in large part on the staff's previous experience with a regulatory guide as the endorsement vehicle for an ASME and ANS voluntary consensus PRA standard for LWRs, as per RG 1.200.

The implementation of this guidance is expected to reduce the need for an in-depth review of the PRA by the NRC staff, allowing 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.

Background

In 1995, the NRC issued a policy statement titled, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement" (Ref. 1). This policy statement encourages the use of PRA in all regulatory matters and states that, "the use of PRA technology should be increased to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach." Additionally, on July 28, 2000, the staff issued SECY-00-0162, "Addressing PRA Quality in Risk-Informed Activities" (Ref. 18), 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), which presents the staff's approach to defining the quality needed for current or anticipated applications, as well as the process for achieving this quality, while allowing risk-informed decisions to be made using currently available methods until all of the necessary guidance documents are developed and implemented. SECY-07-0042, "Status of the Plan for the Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality," dated March 7, 2007 (Ref. 20), provides an update to 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 and reliable. Consequently, the PRA's technical content needs to be complete, correct, and accurate

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and produce 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.[2] PRA acceptability describes the ability of a PRA to support risk-informed regulatory decisionmaking and, for a PRA, is defined in terms of meeting the NRC regulatory position 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. These three aspects each depend on the other to achieve PRA acceptability, as illustrated in Figure 1.

Figure 1. NRC general framework for achieving PRA acceptability

National consensus PRA standards provide one set of minimum requirements that can be met, as endorsed by the staff with exceptions and clarifications, for a PRA to be considered acceptable. (Because these PRA standards use the terms "requirement," "require," and other similar mandatory language, the staff's endorsement, including staff exceptions and clarifications, mirrors this language. However, the use of this language in this RG is not intended to convey a regulatory requirement or suggest that these standards are the only way to meet the statutory and regulatory requirements. The staff will explicitly identify any NRC legal requirements discussed in this RG.) National consensus PRA standards include both technical requirements and process-related requirements such as those related to peer review and PRA configuration control. The PRA peer review process is used to determine whether a PRA meets the requirements provided in the national consensus PRA standard. One acceptable approach for performing a peer review of a PRA is to perform an established, NRC-endorsed peer review process by qualified personnel that documents the results and identifies both strengths and weaknesses of the PRA. When used in support of an application, the use of this RG will reduce the need for an in-depth review of the PRA by NRC reviewers, allowing them to focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application. The acceptability of a PRA is measured against the PRA scope, level of detail, conformance with the NRC regulatory position in Section C of this RG, and representation of the modeled plant.

[2] The term "PRA acceptability" and related phrasings are synonymous with previously used terms such as "PRA quality" and "PRA technical adequacy." The staff uses the term "PRA acceptability" with respect to the scope, level of detail, conformance with PRA elements (i.e., technical adequacy), and plant representation of a PRA as related to the outcome of the NRC staff's review of a given application . For additional information, see DPO-2016-001 (ADAMS Accession No. ML17013A015).

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National consensus PRA standards have been developed by ASME and ANS. In February 2021, ASME and ANS jointly issued the advanced non-LWR consensus PRA standard, ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non- Light Water Reactor Nuclear Power Plants" (Ref. 21), referred to hereafter as the ASME/ANS NLWR PRA standard. The ASME/ANS NLWR PRA standard provides requirements for a comprehensive probabilistic radiological risk assessment (CPRRA) 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). This RG provides the staff endorsement of ASME/ANS NLWR PRA standard. Appendix A to this RG, documents the staff bases for its endorsement, with exceptions and clarifications, of ASME/ANS NLWR PRA standard. The staff's endorsement, for trial use, with exceptions and clarifications, is based on the staff's review of the ASME/ANS NLWR PRA standard against the related regulatory position in Section C of this RG.

A PRA peer review process for NLWR PRA was developed and documented in NEI 20-09, Revision 1, "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard" (Ref. 22), 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. 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" (Ref. 23), as well as other related industry guidance documents that preceded NEI 17-07, Revision 2. The PRA peer review process has been applied by reactor owners' groups and other industry organizations 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 addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis (i.e., a CPRRA). 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 as a means of satisfying the peer review requirements in the ASME/ANS NLWR PRA standard, as endorsed by the NRC in this RG .

Initial licensing application activities generally refer to any one or more of the types of applications shown above. The NRC staff notes that current regulations do not require applicants for 10 CFR Part 50 construction permits or operating licenses to provide PRA-related information, however:

- The Commission's severe accident policy statement (Ref. 4) 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. 5) articulates that all new nuclear power plant designs should meet the Commission's safety goals (Ref. 6).

- There is 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] that may add PRA-related requirements for 10 CFR Part 50 construction permit and operating license applications that will be similar to the existing requirements for Part 52 licenses, certifications, and approvals.

[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

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On January 14, 2019, the President signed the Nuclear Energy Innovation and Modernization Act (NEIMA) into law (Ref. 7). Consistent with Section 103 of NEIMA, the NRC staff has begun efforts 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." The NRC staff notes that the phrase "risk-informed, technology-inclusive" implies that the acceptability of PRA and its results will be an important issue for all reactors (specifically including NLWRs) licensed under the proposed 10 CFR Part 53 for which PRA plays a role in licensing or operation, or both.

The NRC staff endorsement provided in this RG applies to NLWRs that are intended to be installed and operated at a fixed site (stationary reactors), which include (1) reactors that are constructed at a site, and (2) reactors that are constructed at an offsite facility and subsequently transported 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 its 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 desire 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.

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

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

- defining the acceptability of a PRA and its results used in support of an application;
- the NRC's position on national consensus PRA standards, industry PRA peer review process documents, and other related industry documents;
- demonstrating that the PRA and its results used in an application are acceptable; and
- documentation to support a regulatory decision

[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

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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. 24), and Management Directive and Handbook 6.6, "Regulatory Guides," dated May 2, 2016 (Ref. 25).

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

- IAEA Safety Standards Series No. SSG-3, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants," issued 2010 (Ref. 26)

- IAEA Safety Standards Series No. SSG-4, "Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants," issued 2010 (Ref. 27)

Documents Discussed in Staff Regulatory Guidance

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

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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) comprises 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 decisionmaking 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 acceptability of the PRA and its results with respect to the PRA 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.

- 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 captured. 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.[5] Consequently, the PRA should be maintained and upgraded, where necessary, to ensure it represents the as-built and as-operated plant via an

[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 pre-operational stages of the plant lifecycle where 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).

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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. A PRA and its results used to support an application should generally:

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

- Address all internal and external hazards. Seismic events should always be included; other external hazards should also be included if they cannot be screened out with appropriate justification. Appendix B to this RG provides a list of hazards to consider when developing the PRA.

- Address all POSs (e.g., at-power and low-power and shutdown (LPSD)-types of POSs).

- Develop the frequencies of event sequences 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 decisionmaking. 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," (51 FR 28044; August 4, 1986 as corrected and republished at 51 FR 30028; August 21, 1986), are:

- 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 one mile of the site exclusion area boundary. If there are no individuals residing within one mile of the plant boundary, an individual should, for evaluation purposes, be assumed to reside one 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, etc.) 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 one 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 one week of the release) from early phase exposure pathways such as cloudshine,

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groundshine, inhalation, and resuspension inhalation, and during the long-term phase from long-term exposure pathways such as groundshine and resuspension inhalation.

There are several important issues that should be considered when contemplating the development of a PRA that uses risk surrogates to support applications:

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

- Large release frequency (LRF) is used as a risk metric for LWR 10 CFR Part 52 design certification and combined license applications, 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. 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, the staff has not developed a definition of LRF. Staff practice has been to allow Part 52 applicants to define LRF.

- The NLWR PRA standard addresses the development of a PRA that characterizes risk with cumulative risk metrics, but also allows the use of user-defined 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 the effect on the plant. A hazard is a category of similar challenges to plant operations that poses some risk to a facility. For example, internal events 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, where the distinction between these hazard types is defined by the plant boundary in the PRA. The hazard groups addressed in this RG include the following:

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

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- 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. As such, there are a number of internal and external hazards that 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.1.3.6 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.1.3.9 provides additional guidance on the modeling of such hazards. An "other hazards" PRA is performed when other hazards cannot be screened out by a screening analysis. Appendix B to this RG provides a listing of and general description for the internal and external hazards that should be considered during the development of a PRA.

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

C.1.2 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 captured. This minimum level of detail is implicit in the elements comprising the PRA and their associated characteristics and attributes.

For a given PRA element and specific PRA analysis elements, the level of detail modeled in the PRA 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 within the PRA, all technical characteristics and attributes should be addressed. That is, each characteristic and attribute is always addressed, but the degree to which it is addressed may vary.

In general, the level of detail needed in a PRA that supports a risk-informed decision is dependent 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 there are varying levels of available information and operating experience for each of these stages, the submitted PRA will likewise differ in level of detail relative to 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.

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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 provides the list of 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 a complete PRA.

Table 1. PRA Elements

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

The internal events PRA analysis elements listed in Table 1 are the foundational elements needed for a PRA model that addresses all radiological sources, all hazards, all modes of operation, and all levels of analysis. 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. As such, 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 for a given hazard group are provided in the discussion of 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 risk integration PRA element as it relates to LWR PRA acceptability; however, the staff is promulgating a staff 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 PRA Element

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

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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 where 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, 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 re-evaluated and modified with as-operated details. The characteristics and attributes needed to achieve the objectives of a POS analysis PRA element are as follows:

- A set of POSs is identified and characterized during an operating evolution (e.g. full power, LPSD-types of POSs) into distinct operating states where the plant conditions are assumed to be relatively constant.

- The LPSD-types of POSs evolutions are divided into POSs based on the unique impact of plant response 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.

- 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.

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

C.1.3.2 Initiating Event Analysis PRA Element

This section identifies the objectives and the characteristics and attributes of the initiating event analysis PRA element for a 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. One should gain an understanding of the nature of events to promote grouping of events that allows for managing a large number of 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 PRA Element

This section identifies the objectives and the characteristics and attributes of the event sequence analysis PRA element for a 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 course of the accident progression. The characteristics and attributes needed to achieve the objectives of an event sequence analysis PRA element are as follows:

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- 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 plant operating state.

- 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 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 PRA Element

This section identifies the objectives and the characteristics and attributes of the success criteria analysis PRA element for a 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 are dependent on the initiator and the conditions created by the initiator. The computer codes used to perform the analyses for developing the success criteria are validated and verified for both technical integrity and suitability to assess plant conditions of interest for prevention of a release, release in each of the reactor-specific release categories, and they accurately analyze the phenomena of interest. Calculations are performed by personnel who are qualified to perform the types of analyses of interest and are well trained in the use of the codes. 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 are 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 success criteria analysis PRA element.

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

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C.1.3.5 Systems Analysis PRA Element

This section identifies the objectives and the characteristics and attributes of the systems analysis PRA element for a 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 the 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. Human induced security events (e.g. sabotage, malevolent acts) are not included in the scope of considered HFEs. 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 capture both inter-and intra-system dependencies, including support systems, and impacts of dependencies and abnormal environmental.
- The models include both active and passive components and failure modes that impact the functions of the system.
- The models include the common-cause failures, human errors, unavailability resulting from test and maintenance, phenomenological effects, as well as dependencies on POSs.
- The models include the mission times, failure modes associated with system maintenance, component actuation and functionality, and associated human failure events.
- The models include the sources of model uncertainties related to the system analysis.

C.1.3.6 Human Reliability Analysis PRA Element

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

The objectives of the human reliability analysis PRA element are to identify and define the HFEs that can negatively impact normal or emergency plant operation and quantifies 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.

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References such as, but not limited to, NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)," issued April 2005 (Ref. 28), NUREG-1842, "Evaluation of Human Reliability Analysis Methods Against Good Practices," issued September 2006 (Ref.29), and NUREG-2198, "The General Methodology of an Integrated Human Event Analysis System (IDHEAS-G)," issued May 2021 (Ref. 30) 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 a human reliability analysis PRA element are as follows:

- The human reliability analysis 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 human reliability analysis PRA element are identified and analyzed.

C.1.3.7 Data Analysis PRA Element

This section identifies the objectives and the characteristics and attributes of the data analysis PRA element for a 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 to:

- Clearly define the parameter boundaries
- Appropriately group components
- Ensure the parameter data are consistent with parameter definitions
- Comprise 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, actual operating history and experience of the plant when it is of sufficient

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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 PRA 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 for 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 internal flood-specific PRA analysis elements, applicable to all phases leading up to and including the as-built, as-operated plant, are:

- 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 in nature to that for the internal events. Because of its dependence on the internal events model, the internal flood PRA incorporates the elements of regulatory positions C.1.3.1 through C.1.3.7 of this RG, as necessary.

The internal flood 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 form of differences in the flood pathways and water levels, internal flood-induced failure probability of SSCs, the plant response for the LPSD-types of POSs, or a combination of the 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 flood analysis. Flood areas are defined on the basis of physical barriers, mitigation

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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 flood.
- Area definitions are verified through plant walkdowns or via the evaluation of available data and findings of investigations(s) 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.
- 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
    - 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 via the evaluation of available data and findings of investigations(s) 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.

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 is examined (e.g., submergence, spray, pipe whip, and jet impingement). Flood scenarios are developed by examining the potential for propagation and giving credit for flood mitigation. Flood scenarios can be eliminated on the basis of screening criteria. The screening criteria used are well-defined and justified. 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 the different POSs. These

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differences from at-power configurations may account for changes in flood pathways, changes to 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
  - 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 POSs 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 are 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 human reliability analysis to account for performance shaping factors that are attributable to flooding, and (5) quantify internal flood source terms and radiological consequence. 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 PRA 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 provided 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:

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- 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 human reliability analysis
- internal fire event sequence quantification

Internal fire PRA models for at-power- and LPSD-types of POSs are similar in many ways, differing primarily on the relevant operating experience and plant configuration. The internal fire PRA model for a particular POS also relies upon 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 Sections C.1.3.1 through C.1.3.7 of this RG, as necessary.

The objective of internal fire plant boundary definition and partitioning PRA analysis element 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), commonly known as fire areas or compartments. The entire fire PRA is generally organized according to these physical analysis units. 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 capture physical analysis units (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:

- Global analysis boundary captures all plant locations relevant to the internal fire PRA for LPSD-types of POSs.

- PAUs are identified by credited partitioning elements that are capable of substantially confining 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 capture 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 PRA analysis element 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 low power and shutdown 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

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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 physical analysis units. The internal fire PRA model for each POS should be evaluated for the need for different or additional equipment, in particular due to 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, in particular due to 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 internal fire cable selection and location PRA analysis element 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 physical analysis units, 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 internal fire qualitative screening PRA analysis element is to eliminate certain physical analysis units defined in the plant boundary definition and partitioning element that can be shown to be unimportant to fire risk for a POS. These screening criteria should be general qualitative criteria. Those physical analysis units screened out in the internal fire qualitative screening PRA analysis element play no role in the more detailed quantitative assessment. The characteristics and attributes

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needed to achieve the objectives of an internal fire qualitative screening PRA analysis element are as follows:

- Qualitatively screened out physical analysis units 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 internal fire plant response model PRA analysis element is to develop a logic model that represents the plant response following an internal fire. This model is based upon 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 system, structure, and component 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 upon 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 on 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 internal fire scenario selection and analysis PRA analysis element is to define and analyze fire event scenarios that capture the plant fire risk associated with each physical analysis unit. 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 physical analysis units, 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. Data, in particular for LPSD-types of POSs, 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 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).

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- 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 with respect to plant activities which have a bearing on their availability. For example, address dependencies between activities in the plant such as removing a fixed suppression system from service in an area while performing hot work.
- Fire barrier failures should also be addressed under the context of those plant activities leading to the demand of 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 internal fire ignition frequencies PRA analysis element is to estimate the frequencies for 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-type 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 frequencies PRA analysis element are as follows:

- Frequencies are established for ignition sources and consequently for physical analysis units.
- Transient fires should be postulated for all physical analysis units regardless of administrative controls.
- Appropriate justification should be provided to use nonnuclear experience to determine internal fire ignition frequency.

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- 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 internal fire circuit failure analysis PRA analysis element 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 upon 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 are 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 internal fire human reliability analysis PRA analysis element is to identify operator actions and related human failure events (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 human reliability analysis 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.

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

The objective of internal fire event sequence quantification PRA analysis element is to calculate the frequency of the internal fire-induced event sequence (i.e., the fire ignition frequency and the probability of fire damage) and integrate this factor 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 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 pre-established screening criteria for all POSs.

C.1.3.10 Seismic PRA 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.

This section identifies the hazard group-specific PRA analysis element for a seismic PRA for all POSs. The objective for 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 are evaluated for all POSs and may have different characteristics across POSs. The seismic PRA analysis elements for a seismic PRA, applicable to all phases leading up to and including the as-built, as-operated plant, are:

- seismic hazard analysis
- seismic fragility analysis
- seismic plant response analysis

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

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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 form of differences in the seismic capacity of SSCs, the plant response for the LPSD-types of POSs, or both as compared to at-power-types of POSs.

The objective of a seismic hazard analysis PRA analysis element 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. Where no prior study exists, the site-specific probabilistic seismic hazard should be generated. However, in many cases, an existing study can be used to develop a site-specific probabilistic seismic hazard.

In a probabilistic seismic hazard analysis, an essential part of the methodology is the consideration of uncertainties associated with the randomness of events and the state of knowledge (i.e. aleatory and epistemic uncertainties) and typically results in generating 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 earthquakes at the 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.
- Uncertainties are considered in the hazard analysis (in characterizing the seismic sources and the ground motion propagation)

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- properly accounted for
- fully propagated
- allow estimates of fractile hazard curves, median and mean hazard curves
- uniform hazard response spectra

- Spectral shapes used in the seismic PRA are:
  - based on a site-specific evaluation
  - broad-band, smooth spectral shapes for lower-seismicity sites 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 a seismic fragility analysis PRA analysis element 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, 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 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 participate in event sequences for each POS modeled in the seismic PRA systems model
  - describes the basis for screening of high-capacity components

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- Seismic fragility evaluation is performed for SSCs for each POS based on:
    - 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
    - walkdowns or the evaluation of available data and findings of investigations(s) 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
- Plant walkdowns (or the evaluation of available data and findings of investigations(s) 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 POS as applicable, including but not limited to, anchorage, lateral seismic support, and potential systems interactions
- Uncertainties related to the seismic fragility are identified and characterized.

The objective of a seismic plant response analysis PRA analysis element is to determine the plant response to and radiological consequences from 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. The analysis is usually carried out by using the internal events PRA model for each POS as the foundation and unique seismic event related aspects 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:
    - seismic-caused initiating events
    - seismic-induced SSC failures
    - non-seismic-induced unavailabilities

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- 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 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 seismic hazard analysis

  - the seismic fragilities analysis

  - 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 of 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 PRA Element

This section identifies the objectives and the characteristics and attributes for a hazards screening analysis for a 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 employed 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 provides a list of hazards that should be addressed in the PRA. However, in order to help ensure 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 pre-defined 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 re-evaluated 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

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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, as defined in the ASME/ANS standard, to develop a quantitative estimate of risk. Walkdowns of the plant site and plant buildings or the evaluation of available data and findings of investigation(s) of the plant design and operations information are used to confirm assumptions that help form the basis for screening. For some applications, the validity of the assumptions used to screen out a hazard from the PRA may need to be examined and confirmed by the staff 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 pre-defined qualitative screening criteria.
- A detailed screening analysis is performed using pre-defined 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 via the evaluation of available data and findings of investigations(s) 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 PRA 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.

This section identifies the hazard group-specific PRA analysis element for a high wind PRA for all POSs. The objective for 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:

- high wind hazards analysis
- high wind fragility analysis
- high wind plant response analysis

The types of high wind events that should be considered in the analysis are site dependent. These can include tornados and their effects, cyclones, hurricanes, and typhoons, as well as thunderstorms, squall lines, and other weather fronts. It is assumed that the high wind PRA is based on modifications made to an existing, up-to-date internal events, at-power PRA. The technical elements for a high wind PRA are similar to those for a seismic PRA. Because of its dependence on the internal events model, the high wind PRA incorporates the elements of regulatory positions C.1.3.1 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 uses up-to-date databases. Uncertainties in the models and parameter values

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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
  - is reflects recent information
- Uncertainties in the models and parameter values:
  - are properly accounted for
  - are fully propagated
  - allow estimate of mean hazard curve

The objective of a high wind fragility analysis PRA analysis element 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 via the evaluation of available data and findings of investigations(s) of the plant design and operations information for plants that have not started construction or do not have completed enough construction to perform physical walkdowns.
- The uncertainties related to high wind fragility analysis are identified and characterized.

The objective of a high wind plant response analysis PRA analysis element 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 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 internal events PRA model.
- The uncertainties related to high wind plant response analysis PRA analysis element are identified and characterized.

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C.1.3.13 External Flooding PRA 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
- 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 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 form of differences in the flood pathways and water levels, external flood-induced failure probability of SSCs, the plant response for the LPSD-types of POSs, or both as compared to at-power-types of POSs.

The objective of an external flood hazard analysis is to estimate the frequency of external flood at the site using a site-specific probabilistic hazard analysis that incorporates the available recent site-specific information and uses up-to-date databases. Uncertainties in the models and parameter values are properly accounted for and fully propagated to allow the derivation of a mean hazard curve from the family of hazard curves obtained. 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 flood at the site
  - is based on site-specific data or for a justified representative or bounding site, as applicable
  - reflects recent information

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

The objective of an external flood fragility analysis PRA analysis element 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 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 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
  - includes all SSCs that participate 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:
  - a review of plant design documents
  - plant configuration
  - a walkdown or the evaluation of available data and findings of investigations(s) 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
- The uncertainties related to external flood fragility analysis are identified and characterized.

The objective of an external flood plant response analysis PRA analysis element 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:

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- 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 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 of 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 PRA 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:

- other hazards analysis
- other hazards fragility analysis
- other hazards plant response analysis

Screening methods can often be employed 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 D-1 of Appendix D that are not screened out based on a screening and conservative analysis for all modeled POSs and sources of radioactive materials. It should be noted that because of the limited collective

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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 other hazards analysis PRA analysis element is to establish the frequency of occurrence of different intensities of the hazard being analyzed and 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 an other hazard analysis PRA 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 other hazards fragility analysis PRA analysis element is to perform an evaluation to estimate the fragility or vulnerability of an SSC (or their combination) 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 an other hazards fragility analysis PRA analysis element 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(s) 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 other hazards plant response analysis PRA analysis element 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 an other hazards plant response analysis PRA analysis element 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.

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

C.1.3.15 Event Sequence Quantification PRA Element

This section identifies the objectives and the characteristics and attributes of the event sequence quantification analysis PRA element for a 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 for any stage of the plant lifecycle and while ensuring that all risk-significant contributors are represented and understood. The event sequence quantification analysis PRA 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 provide estimates of the frequency of radionuclide releases that could 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 an event sequence quantification analysis PRA element 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 breaking 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 PRA Element

This section identifies the objectives and the characteristics and attributes of the mechanistic source term analysis PRA element for a 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 PRA element is to characterize the radiological release to the environment resulting from each event sequence leading to such 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. Calculations are performed by personnel who are qualified to perform the types of analyses of

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interest and are well-trained in the use of the codes. 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 **PRA 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 a NLWR PRA that addresses all radiological sources, all hazards, all POSs, and all levels of PRA analysis.

This section identifies the specific PRA analysis elements for a radiological consequence analysis performed to support a PRA. 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 pathways. 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 groundwater, are not addressed by these PRA analysis elements. The PRA analysis elements for a radiological consequence analysis are:

- radionuclide release characterization
- site characterization
- meteorological data analysis
- atmospheric transport and diffusion analysis
- protective action analysis
- dosimetry
- health effects analysis
- economic factors
- 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

[6] It should be noted that 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, and their supporting documents) is available.

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(as discussed in Section C.1.3.16 on mechanistic source term analysis). Source term information, developed as discussed in Section C.1.3.16 on mechanistic source term analysis, 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 Section 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 radioisotopes, the time-dependent isotopic release rates to the atmosphere, and the data needed to estimate plume rise.

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

The objective of site characterization PRA analysis element 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 km (50 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 contains 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 PRA analysis element 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 is 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.
- Meteorological data is of acceptable quality and completeness.

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- Meteorological data includes, at a minimum, data on wind speed, wind direction, atmospheric stability, precipitation, and the depth of the atmospheric mixing layer.

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

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

The objective of the atmospheric transport and diffusion analysis PRA analysis element 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 and 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, plume meander, and plume rise) are adequately characterized.

- The deposition of airborne material to the ground by wet and dry deposition, and the resulting depletion of the airborne material with downwind distance, is 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 PRA analysis element 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 contains sufficient information to allow comparison of the postulated protective actions 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.

- 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.

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- 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, 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 is 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.

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

The objective of the health effects analysis PRA analysis element is to perform an assessment of the risk of early or latent health effects (either fatal or non-fatal), 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.

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- 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 PRA analysis element 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 PRA Element

This section identifies the objectives and the characteristics and attributes of the risk integration PRA element for a 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, 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 metric(s) (e.g., radiological doses, health effects to public) or risk surrogates (e.g., large release frequency) are used. Unless justified, relative risk significance criteria, including but not limited to Fussell-Vesely or Birnbaum importance measures, should be used to develop the PRA. The characteristics and attributes needed to achieve the objectives of determining risk-significance criteria are as follows:

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- The analysis defines consequence metric(s) (e.g., person-rem, early fatalities, latent health effects, site boundary dose, quantity of radioactive material release) or risk surrogates (e.g., large release frequency) 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 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 human failure event), 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 sequences 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 in order 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 the objectives of characterizing and quantifying the uncertainties associated with the calculated risk metrics 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

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results is assessed, 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 due to how 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 necessary information so that the results can easily be reproduced and justified. The sources of information used in the PRA also should be referenced and retrievable. The methodology used to perform each aspect of the work is described either through documenting the actual process in the PRA documentation or through reference to existing methodology documents. Sources of uncertainty, both parameter and model uncertainty, are identified and documented and their impact on the results are assessed generally for each technical element. A source of model uncertainty is one that is related to an issue for which there is no consensus approach or model (e.g., choice of data source, success criteria, 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 or, in cases where walkdowns cannot be performed, the documentation describes the

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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 must 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-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 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 decisionmaking 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 with regard to plant procedures and practices
- plant test and maintenance procedures and practices
- engineering aspects of the plant design

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

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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 upon 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
- The SSC's 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)
- 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
- That information needed to reflect planned and typical unplanned tests and maintenance activities and their relationship to the status, timing, and duration of the availability of equipment

The types of 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. This change may occur for the following reasons:

- Operating data may change the availability or reliability of the plant's SSCs.
- Plant design or operation may change.

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- The PRA model may change as a result of improved methods or techniques.

Therefore, to ensure 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 configuration control of computer codes used to support the PRA are maintained.
- 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 and clarifications, that address the scope of the PRA. ASME and ANS have issued the ASME/ANS NLWR PRA standard. The ASME/ANS NLWR PRA standard provides process and technical requirements for a PRA of a 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 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 staff exceptions and clarifications, mirrors this language. The use of this language in this RG is not intended to convey that that compliance with this RG is mandatory, or provides the only way to meet the statutory and regulatory requirements, or that these requirements would be applied to licensees absent their adoption and consistent with the requirements of 10 CFR 50.109, "Backfitting."

Regulatory position C.2.1 of this RG provides the staff position on the use of a national consensus PRA standard to meet regulatory position C.1. To demonstrate acceptability of the PRA, 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 staff with exceptions and clarifications, so that it can be demonstrated that the PRA model is in conformance with regulatory position C.1. Regulatory position C.2.2 of this RG provides the staff position on the use of PRA peer reviews to this effect, including staff endorsement with exceptions and clarifications, of related industry PRA peer review guidance. When the peer review accounts for regulatory position C.2.2 and the PRA is assessed against a national consensus PRA standard consistent with regulatory positions C.1 and C.2.1, including staff exceptions and clarifications, this represents an acceptable peer review process. The PRA is considered to be acceptable by the NRC staff for supporting that the application based on the results of

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the peer review, resolution of the Fact and Observation (F&O), and how the PRA conforms to the requirements in the national consensus PRA standards.

C.2.1 National Consensus Probabilistic Risk Assessment Standards

National consensus PRA standards provide requirements for what is needed in order to develop 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, Capability Category I and Capability Category II, which are used to distinguish between greater and lesser scopes, levels of detail, plant representation, and realism needed for a given technical requirement. Where a Supporting Requirement provides a different requirement for each Capability Category, the Capability Category I requirement generally fosters identification of the most risk-significant event sequences at a functional or systemic level. The Capability Category II requirement fosters the development of a realistic assessment of risk. The capability category 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., Capability Category I) to the minimum capability needed to meet current good practice (i.e., state-of-practice) for each PRA element (i.e., Capability Category II). Further, the capability category that needs to be met for each technical requirement is dependent on the application. In general, the staff anticipates that meeting Capability Category II should result in an acceptable scope, level of detail, and realism for the majority of applications. However, for some applications, Capability Category I may be acceptable for some requirements.

The requirements in an ASME/ANS PRA standard are either process-related or are technical in nature. 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 straight forward 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 requirement may be considered to be met by the staff. 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 capture the objective of the PRA element. HLRs are defined in general terms, should be met regardless of the capability category, and accommodate different approaches. SRs are defined for each HLR and are the 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.

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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 and clarifications, have been met. An acceptable peer review approach is one that is performed according to an established process and by qualified personnel, documents the results, and identifies both strengths and weaknesses of the PRA. The ASME/ANS 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, "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard," as one acceptable approach for determining whether a PRA meets the requirements provided in the ASME/ANS NLWR PRA standard. In addition to the guidance provided in NEI 20-09, Revision 1, the ASME/ANS NLWR PRA standard also provides 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 and clarifications, 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 set forth 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 provided in NEI 20-09, Revision 1, is considered to be acceptable for a peer review performed for a PRA representing any stage of plant lifecycle 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. Additional guidance regarding PRA peer review is provided below.

As part of an application, an applicant or licensee should describe the measures it has taken to assure that the design-specific or plant-specific PRA is acceptable for the application. 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.

Peer review of the PRA is not required for a DC, SDA, ML, CP, COL, or OL application. However, the staff suggests that a pre-application peer review of the PRA be performed to determine the technical acceptability of the PRA against the ASME/ANS NLWR PRA standard, as endorsed by the NRC. A peer review is one acceptable and effective way to identify strengths and weaknesses in the PRA, which is an essential step in establishing confidence in the PRA results and risk insights.

When performed prior to the application, the peer review provides findings and observations regarding PRA completeness and acceptability, including consideration of the scope, level of detail,

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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 provide the staff with an appropriate level of 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 have a significant impact on 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 upon 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 they reflect the as-designed, as-to-be-built, as-to-be-operated plant, or the as-built and as-operated plant. Additionally, independent walkdowns, if achievable, are performed by the peer reviewers 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 peer review 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 peer review team should have technical expertise in the PRA elements they review, including experience in the specific methods used to develop PRA 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 should have familiarity with the general design and operating philosophy based on the design and operating guidance available for that stage of the plant lifecycle.

Chapter 4 of NEI 20-09, Revision 1, and Section 6.2 of the ASME/ANS NLWR PRA standard provide specific peer review team qualifications that should be met. 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 possess the following attributes which are listed in section 4.4 of NEI 20-09:

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

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

- Knowledgeable of the specific reactor technology and its design used for the development of the PRA under review

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

The staff agrees with the discussion in NEI 20-09, Revision 1, that, due to 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 plant operating states, 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. This includes consideration of:

- 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 development of a list of issues to be addressed in the PRA
- Documentation of the review conclusions.

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

- Preparatory activities
- Offsite review
- The onsite consensus review
- 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 Capability Categories (CCs) to each of the SRs of the various PRA elements of the PRA standard or to judge whether the PRA meets the SRs in the ASME/ANS NLWR PRA standard within the scope of the review. A summary of the SR review is then provided for each HLR. For each CC, the SRs define the minimum requirements necessary to meet that CC. Some of the SR action statements apply to only one CC while others span across 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.

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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

- 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 peer 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 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 NDM should be peer reviewed to determine the NDM's acceptability. The peer review of an NDM assesses whether the NDM meets some established set of technical requirements and, consequently, can be used to support the PRA. After an NDM has been successfully peer reviewed, the NDM's implementation into 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 the guidance 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 NDM 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 an adequate level of control over the use of an NDM. If open F&Os from an NDM peer review cannot be successfully closed via an NRC-endorsed closure process, the NDM could be submitted to the NRC to determine the acceptability of the NDM.

The staff recognizes that for certain types of NDMs (e.g., fundamentally novel methods or technologies) some direct review from NRC staff may be warranted. Submitted applications that use NDMs with open F&Os related to the NDM are subject to review by the NRC to determine the acceptability of the NDM, its implementation in the PRA, and its potential impact on the application. The peer review of an NDM should meet certain requirements specific to that type of peer review. An acceptable set of requirements against which the acceptability of an NDM can be assessed is listed below:

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- 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 non-numeric in nature) is 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, and given the purpose and scope of the NDM.
- The documentation of the NDM provides traceability of the work and facilitates incorporation of the NDM in a PRA model.

C.3 Demonstrating the Acceptability of a PRA Used to Support an Application

This section of the RG provides guidance to applicants and licensees on an approach acceptable to the NRC staff on how 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:

- Describe the PRA's scope, 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; and
- 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 position in this RG, and relevant application-specific regulatory guidance.

C.3.1 PRA 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 the application-specific requirements. Guidance on meeting such requirements is provided in application-specific guidance documents.

For plants in the pre-operational stages of the plant lifecycle, 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. The PRA should always reflect the best available information for the plant, when used for risk informed decision-making. For most applications, an applicant or holder of a license, certification, or permit should address all radiological sources, all hazards, all plant operating states, and all levels of analysis, as discussed in Section C.1.1 of this RG. The staff will assess the appropriateness of the justification for any deviations from this scope.

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C.3.2 Development and Use of an Acceptable PRA

The staff positions in sections C.1 through C.1.4 represent the minimum capability the staff have determined 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 is to use an NRC-endorsed national consensus standard during the development of the PRA and to have the PRA peer reviewed via 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 staff in Appendix A to this RG, the staff positions in Sections 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 and clarifications 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 the purpose for which it was intended. If the PRA is demonstrated to have met the requirements of these documents, with attention paid to the NRC's exceptions and clarifications, it can be assumed that the analysis is technically correct. As such, 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 are such that no changes occur in the risk contributors.

As discussed in Section C.2.2.1 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 and clarifications, have been met, which includes assessing the appropriateness of assumptions and approximations used in the PRA. This helps assure 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 they reflect 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 lifecycle. The results of a peer review should be used to help establish the assurance 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 certain 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. 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

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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 for which 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 for which it is proposed 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 used to support 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 that it correctly reflects the dependencies amongst systems, components, and operator actions, and (3) the probabilities and frequencies used are estimated consistently with the definitions of the corresponding events in the PRA logic model 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 use 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 Section 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

The applicability of acceptance criteria or guidelines for a given application is provided in related application-specific guidance documents. Such guidance documents should address the PRA results needed to compare against the acceptance criteria or guidelines and how the comparison should be performed. As such, 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 the policy statement titled, “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 Section C.1.1 of this RG. The Commission’s Safety Goals policy statement expresses the Commission’s views on the level of risks to the public health and safety that the nuclear industry should strive to meet for nuclear power plants. As such, if the safety goals and QHOs are

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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 PRA 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. As such, 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 PRA and its results are used to support the application.

During the course of developing an application, the applicant or holder of a license, certificate, or permit develops documentation of 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 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. Guidance on how to meet documentation submittal requirements for a specific application is provided in application-specific guidance. Section C.4.2 details aspects of submittal guidance that should be met, independent of any specific application.

C.4.1 Archival PRA Documentation

Certain 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 and for all applications. As part of achieving these characteristics and attributes, a detailed description of the following should be developed for archival PRA documentation:

- The process used to determine the acceptability of the PRA including a description of how the staff position in this RG is met, such as whether a national consensus standard was used and whether the technical requirements in that standard are met.
    - If a national consensus standard was used as part of demonstrating PRA acceptability, the documentation should demonstrate that the PRA was developed consistently with that national consensus standard, as endorsed by the NRC in this RG.
    - If a national consensus standard was not used or was used in part to demonstrate PRA acceptability, justification should be developed for each requirement that is not met regarding 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 methodology used to assess the risk of the application, to include details of how the risk was quantified and identification and justification of all assumptions and approximations used to develop or evaluate the PRA

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- SSCs, operator actions, and plant operational characteristics affected by the application, including the cause-effect relationships amongst 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 aforementioned 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

C.4.2 Submittal PRA Documentation

In addition to the characteristics and attributes of archival PRA documentation described in C.4.1, there are characteristics and attributes of submittal PRA documentation that are likewise fundamental to the staff's assessment of PRA acceptability for a given application and that should be achieved during the creation of that documentation by an applicant or holder of a license, certification, or permit and for any application. As part of achieving these characteristics and attributes, a detailed description of the following should be developed as part of the submittal PRA documentation:

- 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:

    - This should include the identification of permanent plant changes (such as design or operational practices) that have an impact on the PRA but that have yet to be represented in the PRA and justification for why the change does not impact the PRA results used to support the application. This justification should be in the form of a sensitivity study that demonstrates the event sequences or contributors significant to the application were not adversely impacted.

- The appropriateness of key assumptions and approximations and sensitivity studies thereof relevant to the results used in the application (e.g., self-assessments, peer reviews). A peer review of planned modifications should clearly identify and describe the plant modifications and design changes that are modeled in the PRA but are not completed at the time of submittal.

- The appropriateness of a given portion of the PRA that meets a capability category lower than deemed required for the application under consideration.

- The appropriateness of PRA model upgrades, including the use of NDMs, for the application under consideration.

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- For NDMs, this should also include a discussion of 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.

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

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

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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, 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," (Ref. 31) 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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REFERENCES [7]

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

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

3. CFR, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Part 52, Chapter 1, Title 10, "Energy." CFR, Subpart A, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants," Part 52, Chapter 1, Title 10, "Energy."

4. U.S. Nuclear Regulatory Commission, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," 50 FR 32138, August 8, 1985.

5. U.S. Nuclear Regulatory Commission, "Policy Statement on the Regulation of Advanced Reactors," 73 FR 60612, October 14, 2008.

6. U.S. Nuclear Regulatory Commission, "Safety Goals for the Operation of Nuclear mmgPower Plants," 51 FR 28044, August 4, 1986 as corrected and republished at 51 FR 30028, August 21, 1986.

7. U.S. Congress, Public Law 115-439, "Nuclear Energy Innovation and Modernization Act," January 2019.

8. U.S. Nuclear Regulatory Commission (NRC), NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," Chapter 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Washington, DC.

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

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

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

[7] 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.

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

13. NRC RG 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities,” Washington, DC.

14. NRC, RG 1.206, Rev. 1, “Applications for Nuclear Power Plants,” Washington, DC.

15. U.S. Nuclear Regulatory Commission, 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,” Washington, D.C.

16. NRC, DC/COL-ISG-028, “Interim Staff Guidance on Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application,” Washington, DC.

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

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

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

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

21. American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard ASME/ANS RA-S-1.4-2020, “Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants,” January 2021.

22. Nuclear Energy Institute, NEI 20-09, “Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard,” January 2021.

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

24. NRC, “Nuclear Regulatory Commission International Policy Statement,” Federal Register, Vol. 79, No. 132, July 10, 2014, pp. 39415–39418.

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

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26. International Atomic Energy Agency (IAEA) Safety Standard Series No. SSG-3, "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants," Vienna, Austria, 2010. [8]

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

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

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

30. NRC, "The General Methodology of An Integrated Human Event Analysis System (IDHEAS-G)", NUREG-2198, Washington, DC, May 2021 (ADAMS Accession No. ML21127A272).

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

32. NRC, SRM-SECY-90-16, "SECY-90-16 - Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," Washington, DC, June 26, 1990.

33. NRC, SECY-13-0029, "History of the Use and Consideration of the Large Release Frequency Metric by the U.S. Nuclear Regulatory Commission," Washington, DC, 22, 2013.

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

APPENDIX A

NRC REGULATORY POSITION ON ASME/ANS RA-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 "states the requirement for probabilistic risk assessment (PRAs) use to support risk-informed decisions for advanced non-light water reactor (non-LWR) 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 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-1.4-2021is categorized as "no objection," "no objection with clarification," or "no objection subject to the following qualification," and defined as follows:

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

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

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

The ASME/ANS RA-S-1.4-2021 Standard 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
- Mechanistic Source Term Analysis
- Radiological Consequence Analysis
- Risk Integration

Tables A-1 through A-22 provides the staff's position on each of the requirements of the standard. A discussion of the staff's concern (issue) and the staff proposed resolution is provided. In the proposed staff resolution, the staff clarification or qualification to the requirement is indicated in either

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bolded text (i.e., **bold**) or strikeout text (i.e., ~~strikeout~~); that is, the necessary additions or deletions to the requirement (as written in the ASME/ANS RA-S-1.4-2021 PRA standard) have been provided 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.

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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
Global			
General	The phrase "advanced non-LWR" is used throughout the standard but is not defined.	Clarification	*The standard may be applied to any 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.*
Nonmandatory Appendices	Nonmandatory appendices providing explanatory material or examples may be acceptable, however, the staff has not reviewed these.	Clarification	*For nonmandatory appendices: the staff will only provide a staff position for the notes associated with the technical requirement. No staff position is provided for nonmandatory appendices providing explanatory material or specific examples.*
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			

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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
Feasibility	In the context of operator actions, several high-level requirements and supporting requirements refer to feasibility; however, this term is not defined.	Qualification	*Add the following definitions*: (1) **Feasibility assessment – the qualitative consideration of whether the operator action is go/no-go, considering several performance shaping factors.** (2) **Feasible – an operator action that can be credited in a PRA model if the action has met all the feasibility assessment criteria (see supporting requirement HR-H2**)
Skill of the craft	The definition of "skill of the craft" is inadequately detailed and is inconsistent with that given in NUREG-1921.	Qualification	*The following definition of "skill of the craft" should be used:* "**Actions that one can assume that trained staff would be able to perform without written procedures (e.g., simple tasks such as turning a switch or opening a manual valve as opposed to a series of sequential actions or set of actions that need to be coordinated.**"
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	----------

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 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			
Section text	----------	No objection	----------
Section 4.2.6			
Section text	----------	No objection	----------
Section 4.2.7			
Section text	----------	No objection	----------

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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	Limiting the CC-I requirement for POS-A1 only to at-power plant evolutions potentially excludes a significant risk contributor as low-power and shutdown-types of POSs have been shown to have a comparable risk in some cases to at-power POSs. As such, the scope of the CC-I requirement should be the same as the scope of the CC-II requirement to avoid excluding potentially significant contributors to risk.	Qualification	CC-I ~~IDENTIFY a representative set of plant evolutions to be analyzed.~~ ~~INCLUDE, at a minimum, plant evolutions from at-power operations.~~ ~~See Note POS-N-1, POS-N-2, POS-N-3, POS-N-4~~ CC-I and CC-II **IDENTIFY a representative set of plant evolutions to be analyzed, including refueling outages, other controlled shutdowns, and forced outages.** **See Note POS-N-3**
POS-A2 through POS-A7	----------	No objection	----------
POS-A8	Additional requirements are needed for CC-I and CC-II,	Qualification	CC-I and CC-II For PRAs performed during the pre-operational stage, INTERVIEW

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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
	operations personnel supporting a licensing application need to confirm that the selection of plant operating states correctly represents the as-designed and as-intended-to-operated plant		knowledgeable engineering **and operations personnel** to confirm that the selection of ~~plant operating states~~ **POSs** correctly represents the as-designed, and as-intended-to-operated plant.
POS-A9	----------	No objection	----------
POS-A10	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 that appropriateness of the POS definition. Without such consideration, POS definitions may not be complete.	Qualification	CC-I and CC-II REVIEW the plant conditions defined for each plant operating state to ENSURE that the plant operating state definition remains sufficient for those hazard groups to do the following: (a) support the selection of initiating events, the justification of success criteria, plant operating states frequency and duration parameters, the evaluations of HFEs, the accounting for planned equipment outages, and the quantification of event sequence frequencies; (b) provide a finite number of sets of plant conditions for peer reviews **(c) account for changing plant conditions that may impair or change the effectiveness of radionuclide transport barriers, affect propagation pathways, or modify fragilities of SSCs to ensure that appropriateness of the POS definition.**
POS-A11 through POS-A13	----------	No objection	----------
Table 4.3.1.1-3			
POS-B1	Omitting the condition to ensure that the POS grouping does not	Qualification	CC-I GROUP plant evolutions into a set of representative evolutions. ENSURE that

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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
	impact risk-significant event sequences could significantly impact the results and insights from the PRA. As such, a new requirement is needed for CCI to reflect as much.		(a) the evolutions within a group can be considered 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	----------
Section 4.3.1.2.4			
Section text	----------	No objection	----------
Nonmandatory Appendix POS: Notes and Explanatory Material for Plant Operating State Analysis			
Section POS.1			
Heading text	----------	No objection	----------
Table POS-1			
POS-N-1	----------	No objection	----------

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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
POS-N-2	All stages of the licensing process should address low power and shutdown-types of evolutions	Clarification	~~Early pre-operational stage PRAs are typically limited to at-power PRAs only.~~ **All stages of the licensing process should address low power and shutdown-types of evolutions**
POS-N-3	----------	No objection	----------
POS-N-4	All stages of the licensing process should address low power and shutdown-types of evolutions	Clarification	~~Depending on the application, the evolution to be addressed may range from at-power only to all plant operating states outage types.~~ **All stages of the licensing process should address low power and shutdown-types of evolutions.**
POS-N-5 through POS-N-10	----------	No objection	----------
POS-N-11	Ensure that this note is pertains to only time-dependent risk profile of plant configurations for a plant- or design-specific evolution.	Clarification	...When documenting the Plant Operating State Analysis for **time-dependent risk profiles**, it is not necessary to explicitly list each of the plant operating states and the specific values of the selected plant conditions in an exhaustive table of all plant operating states .
POS-N-12 through POS-N-20	----------	No objection	----------
POS-N-21	For licensing reviews, the key attributes of the plant conditions for each POS should be clearly documented in a format (e.g. table, chart) to facilitate understanding of the NLWR PRA results by an independent reviewer.	Qualification	The set of plant operating states is the organizing structure for the definition of event sequences modeled in the PRA. **The key attributes of the plant conditions for each POS should be clearly documented in a format (e.g. table, chart) to facilitate understanding of the NLWR PRA results by an independent reviewer.** See POS-D1

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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-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	----------
Section 4.3.2.3			
Section text	----------	No objection	----------

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

Index No.	Issue	Position	Resolution
Nonmandatory Appendix IE: Notes and Explanatory Material for Initiating Events Analysis			
Section IE.1			
Heading text	----------	No objection	----------
Table IE-1			
IE-N-1 through IE-N-35	----------	No objection	----------

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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	----------
Nonmandatory Appendix ES: Notes and Explanatory Material for Event Sequence Analysis			
Section ES.1			
Heading text	----------	No objection	----------
Table ES-1			
ES-N-1 through ES-N-17	----------	No objection	----------

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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	----------
Nonmandatory Appendix SC: Notes and Explanatory Material for Success Criteria Analysis			
Section SC.1			
Heading text	----------	No objection	----------
Table SC-1			
SC-N-1 through SC-N-12	----------	No objection	----------

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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-A30	----------	No objection	----------
SY-A31	There are no commonly used analysis methods for recovery in the sense of repair, other than use of actuarial data.	Clarification	…is justified through an adequate analysis or examination of data **collected in accordance with DA-C20.** ~~(See DA-C20.)~~
SY-A32 and SY-A33	----------	No objection	----------
Table 4.3.5.1-3			
SY-B1 through SY-B11	----------	No objection	----------
SY-B12	This SR is applicable to operating plants.	Clarification	… See Note ~~SY-N-4~~ SY-N-24, SY-N-25.
SY-B13 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			

Pre-decisional

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

Index No.	Issue	Position	Resolution
Section text	----------	No objection	----------
Section 4.3.5.2.2			
Section text	----------	No objection	----------
Section 4.3.5.2.3			
Section text	----------	No objection	----------
Nonmandatory Appendix SY: Notes and Explanatory Material for Systems Analysis			
Section SY.1			
Heading text	----------	No objection	----------
Table SY-1			
SY-N-1	Use of the term "unscreened hazards" suggests that such hazards may have been screened out and subsequently screened back into the PRA as though the act of screening had been undone. However, this term is interpreted as meaning hazards that were that were included in the PRA for consideration and evaluation (i.e., screened in).	Clarification	...If ~~there are~~ other ~~unscreened~~ hazards **are included for consideration and evaluation** ~~added to~~ **in** the **PRA** model, there may be additional SSCs that are added to the scope of the Systems Analysis.
SY-N-2 through SY-N-3	----------	No objection	----------
SY-N-4	This note does not apply to SY-B12.	Clarification	This SR is not applicable to operating plants. See SY-A4, SY-A6, SY-A11, SY-A13, SY-A22, SY-A26, SY-A33, SY-B10, ~~SY-B12~~, SY-B17, SY-C3
SY-N-5	----------	No objection	----------

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

Index No.	Issue	Position	Resolution
SY-N-6	This note expands on requirements that are specific for operating reactors, therefore, the requirements would not apply (at all) to PRAs performed during pre-operational stage.	Clarification	This SR is ~~likely~~ not applicable to PRAs performed during the pre-operational stage.
SY-N-7 through SY-N-26	----------	No objection	----------

Pre-decisional

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 activities" suggests that such activities may have been screened out and subsequently screened back into the PRA as though the act of screening had been undone. However, this term is interpreted as meaning activities that were 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 **included for consideration and evaluation in the PRA**;
Table 4.3.6.1-1			
HLR-HR-A through HLR-HR-D	----------	No objection	----------
HLR-HR-E	The scope of high-level requirement (HLR) HR-E does not include errors of commission. See HR-E4 in this table for more details about the basis for this issue.	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**.
HLR-HR-F through HLR-HR-I	----------	No objection	----------
Table 4.3.6.1-2			
HR-A1 through HR-A10	----------	No objection	----------

Predecisional

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 4.3.6.1-3			
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			
HR-D1 through HR-D3	----------	No objection	----------
HR-D4	The phrase "when available" is inappropriate for an operating plant. Add examples for what is meant by quality in item (b) of the CC-II requirement.	Clarification	CC-II For each detailed HEP assessment, INCLUDE in the evaluation process the following plant- or design-specific relevant information ~~when available~~: (b) the quality of the human-machine interface **(e.g., adherence to human factors guidelines [NUREG-0700, Revision 3] and results of any quantitative evaluations of performance per functional requirements)**, including …
HR-D5	----------	No objection	----------
HR-D6	The phrase "if available" is inappropriate for an operating plant.	Clarification	For operating plants, if recovery of pre-initiator errors is credited, USE the following information~~, if available,~~ to assess the potential...
HR-D7	----------	No objection	----------
HR-D8	The conservative estimate or mean value of the HEP whose uncertainty is being characterized should be provided. Also, the SR that requires the conservative estimate or mean HEP value should be linked to HR-D8.	Clarification	CC-I CHARACTERIZE the uncertainty for the HEPs. **PROVIDE the conservative estimate or mean value of the HEP that results from HR-D2.** CC-II For each risk-significant HFE**s**, PROVIDE a probabilistic representation of the uncertainty of the calculated HEPs**,**

Pre-decisional

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

Index No.	Issue	Position	Resolution
			including the mean value that results from HR-D2. For HFEs that are not risk-significant, ~~CHARACTERIZE the uncertainty~~ **ENSURE the requirement for CC-I is met**.
HR-D9 and HR-D10	----------	No objection	----------
Table 4.3.6.1-6			
HR-E1 through HR-E3	----------	No objection	----------
HR-E4	HR-E4 does not include errors of commission (EOC). EOCs should be included in the advanced non-light water reactor (LWR) PRA standard for the following reasons: (1) the significant amount of experience in operating LWRs facilitates a consensus between NRC and industry to exclude EOCs from the LWR Level 1/large, early release frequency (LERF) PRA standard; however, there is very little (if any) advanced non-LWR operating experience to allow the consensus to exclude EOCs from the advanced non-LWR PRA standard; (2) it is expected that advanced non-LWRs would rely less on human actions than LWRs, which implies that EOCs would play	Qualification	*Add the following to item to HR-E4*: "**(c) those well-intended actions performed by control room staff that disable a system, sub-system, or component needed in an event scenario.**"

Pre-decisional

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

Index No.	Issue	Position	Resolution
	a more important role in advanced non-LWR PRAs than in LWR Level 1/LERF PRAs; and (3) given that (a) the scope of the advanced non-LWR PRA standard covers what in the LWR world is known as Level 2 PRA and (b) there is no consensus about EOCs in Level 2 PRA, the developers of PRAs for advanced non-LWRs should demonstrate that EOCs are not an issue before eliminating them from consideration.		
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	Two issues with HR-G1: (1) Feasibility assessment is a continuous step in the HRA process that can be performed even during HRA quantification. The NRC staff proposes to make the feasibility criteria in HR-H2 applicable to post-initiator HFEs because non-LWRs lack the operating experience	Qualification	CC-I **ASSESS the feasibility of the HFEs before assigning the final HEPs 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 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. If the HFE is not feasible, ASSIGN an HEP of 1.0 or DO**

Pre-decisional

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

Index No.	Issue	Position	Resolution
	currently available in LWRs and the non-LWR PRA standard scope goes beyond the scope of the LWR Level 1/LERF PRA standard. Further, ESQ-C7 requires that human actions be feasible in order to use their respective HEPs in event sequence quantification. (2) The following language in the CC-I requirements "that survive initial quantification" is too prescriptive because it implies an initial quantification should be performed (that is not a requirement) and there is no information provided in the Standard about the requirements of initial quantification.		**NOT CREDIT the HFE in the PRA. 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.
HR-G2 and HR-G3	----------	No objection	----------
HR-G4	In item (d) of the CC-II requirement, clarify that "clarity" refers to the meaning of the cues/indications. Also, the CC-II requirement does not consider communication. Communication is an important performance shaping factor for actions performed outside the main control room (e.g., in	Clarification	<u>CC-II</u> (d) degree of clarity of the cues/indications **in supporting the detection, diagnosis, decision-making, and action execution given the plant-specific and event scenario-specific context** **(p) communication among personnel in the same team and in different teams.**

Pre-decisional

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

Index No.	Issue	Position	Resolution
	fire scenarios) and that require complex coordination, such as the use of FLEX equipment.		
HR-G5	----------	No objection	----------
HR-G6	Supporting requirements concerning the use of thermal-hydraulic analyses/codes should be cross-referenced.	Clarification	CC-I For the time available to complete actions […], USE applicable generic studies (e.g., thermal-hydraulic analysis for similar plants). **(See SC-B4)** CC-II For the time available to complete actions […], USE design- or plant-specific evaluations, appropriate realistic thermal-hydraulic analysis from similar plants […]. **(See SC-B4)**
HR-G7 through HR-G13	----------	No objection	----------
HR-G14	Consistent with the staff position in HR-D8, the point estimate or mean value of the HEP whose uncertainty is being characterized should be provided.	Clarification	CC-I …CHARACTERIZE the uncertainty for the calculated HEPs. **PROVIDE the calculated point estimate HEP**. CC-II …PROVIDE a probabilistic representation of the uncertainty of the calculated HEPs**, including the calculated mean values**. For the HFEs that are not risk-significant, ENSURE the requirement for CC-I is met.
HR-G15 and HR-G16	----------	No objection	----------
Table 4.3.6.1-9			
HR-H1	----------	No objection	----------
HR-H2	Consistent with the position on HR-G1, two feasibility criteria need to be added to HR-H2 because non-LWRs lack the	Qualification	*Add the following two feasibility criteria:* **(f) there is a sufficient plan for command and control;** **(g) there is a sufficient plan for communications.**

Predecisional

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

Index No.	Issue	Position	Resolution
	operating experience currently available in LWRs and the non-LWR PRA standard scope goes beyond the scope of the LWR Level 1/LERF PRA standard.		
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	----------
Nonmandatory Appendix HR: Notes and Explanatory Material for Human Reliability Analysis			
Section HR.1			
Heading text	----------	No objection	----------
Table HR-1			
HR-N1 through HR-N24	----------	No objection	----------

Preliminary

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-6	----------	No objection	----------
Table 4.3.1-3			
DA-B1 through DA-B2	----------	No objection	----------
Table 4.3.7.1-4			
DA-C1 through DA-C19	----------	No objection	----------
DA-C20	This SR provides a justification for crediting equipment repair (SY-A31). As written, it could be interpreted as allowing plant-specific data to be discounted in favor of industry data. In reality, for such components as pumps, plant-specific data is likely to be insufficient and a broader base is necessary.	Qualification	…IDENTIFY instances of plant-specific experience ~~or~~ **and, when that is insufficient to estimate failure to repair consistent with DA-D10,** applicable industry experience and for each repair, COLLECT…
DA-C21 through DA-C26	----------	No objection	----------
Table 4.3.7.1-5			

Pre-decisional

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

Index No.	Issue	Position	Resolution
DA-D1 through DA-D10	----------	No objection	----------
Table 4.3.7.1-6			
DA-E1 through DA-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	----------
Nonmandatory Appendix DA: Notes and Explanatory Material for Data Analysis			
Section DA.1			
Heading text	----------	No objection	----------
Table DA-1			
DA-N-1 through DA-N-31	----------	No objection	----------

Pre-decisional

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	This standard uses a term of investigation in SRs, replacing walkdown. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development.	Clarification	EVALUATE the Internal Flood Plant Partitioning for the as-built, as-operated or as designed, as-intended-to-operate plant conditions via **walkdowns(s) or, for PRAs performed during pre-operational phase,** investigation(s) depending…
FLPP-B7 through FLPP-B8	----------	No objection	----------
Table 4.3.8.1-4			
FLPP-C1	This standard uses a term of investigation in SRs, replacing walkdown. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development.	Clarification	… *(b)* the general nature and key or unique features of the partitioning elements that define each flood area; *(c)* any **walkdowns(s) or, for PRAs performed during pre-operational phase,** investigation(s) performed in support of the plant partitioning; …

Pre-decisional

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

Index No.	Issue	Position	Resolution
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	Other fluid sources besides water should also be considered.	Clarification	IDENTIFY the potential flood sources that include water, steam, **and other liquids (e .g., lubricating oil , fuel oil)**.
FLSO-A3 through FLSO-A6	----------	No objection	----------
FLSO-A7	This standard uses a term of investigation in SRs, replacing walkdown. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development.	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(s) or, for PRAs performed during pre-operational 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	----------

Pre-decisional

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 4.3.8.3-1			
HLR-FLSN-A and HLR-FLSN-B	----------	No objection	----------
Table 4.3.8.3-2			
FLSN-A1 through FLSN-A18	----------	No objection	----------
FLSN-A19	This standard uses a term of investigation in SRs, replacing walkdown. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development.	Clarification	EVALUATE the accuracy of information collected from plant information sources for the as-designed, or as-built, and as-operated or as-intended-to-operate plant conditions via **walkdowns(s) or, for PRAs performed during pre-operational phase,** investigation(s) depending…
FLSN-A20 through FLSN-A21	----------	No objection	----------
Table 4.3.8.3-3			
FLSN-B1	This standard uses a term of investigation in SRs, replacing walkdown. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development.	Clarification	… *(f)* calculations or other analyses used to support or refine the flooding evaluation; *(g)* **walkdowns(s) or, for PRAs performed during pre-operational phase,** investigation(s) performed; …
FLSN-B2 through FLSN-B3	----------	No objection	----------
Section 4.3.8.4			
Section text	----------	No objection	----------
Table 4.3.8.4-1			

Pre-decisional

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

Index No.	Issue	Position	Resolution
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	The frequency of equipment failure-induced floods for each POS for temporary alignments need to be considered.	Qualification	… (f) impact of POS changes within the scope of the PRA on flood-induced initiating events. **(g) impact of temporary alignments on the frequency of equipment failure-induced floods for each POS.** …
FLEV-C2 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			

Pre-decisional

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

Index No.	Issue	Position	Resolution
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 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 4.3.8.7-1			
Section text	----------	No objection	----------
Table 4.3.8.7-1			

Pre-decisional

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

Index No.	Issue	Position	Resolution
HLR-FLESQ-A through HLR-FLESQ-F	----------	No objection	----------
Table 4.3.8.7-2			
FLESQ-A1 through FLESQ-A5	----------	No objection	----------
FLESQ-A6	This standard uses a term of investigation in SRs, replacing walkdown. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development.	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(s) or, for PRAs performed during pre-operational phase,** investigation(s):…
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			

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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	----------
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	----------
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	----------
Nonmandatory Appendix FL: Notes and Explanatory Material for Internal Flood PRA			
Section FL.1			
Heading text	----------	No objection	----------
Table FL-1			
FL-N-1 trough FL-N-30	----------	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			

Pre-decisional

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	----------

Pre-decisional

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 4.3.9.5-1			
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	----------

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Index No.	Issue	Position	Resolution
Table 4.3.9.6-6			
FSS-E1 through FSS-E5	----------	No objection	----------
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-A3	----------	No objection	----------
FIGN-A4	Reference to F-N-8 is incorrect since this SR is for operating plants. Note F-N-6 should be referenced, rather than F-N-8	Clarification	… See Note **F-N-6,** ~~F-N-8,~~ F-N-56
FIGN-A5 through FIGN-A12	----------	No objection	----------
Table 4.3.9.7-3			
FIGN-B1 through FIGN-B3	----------	No objection	----------

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Index No.	Issue	Position	Resolution
Section4.3.9.8			
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			
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	----------

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Index No.	Issue	Position	Resolution
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			
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			

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Index No.	Issue	Position	Resolution
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			
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	----------
Nonmandatory Appendix F: Notes and Explanatory Material for Internal Fire PRA			
Section F.1			
Heading text	----------	No objection	----------
Table F-1			
F-N-1 through F-N-67	----------	No objection	----------

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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-B4	----------	No objection	----------
SHA-B5	SHA-B5 does not include consideration of (1) the use of an existing probabilistic SHA for a site and, (2) the impact of an updated catalog on the use of the existing probabilistic SHA. Given the likelihood of using an existing site as the bounding site (see SHA-A1), the considerations identified above are warranted.	Qualification	*Add the following to SHA-B5*: **If an existing probabilistic SHA is used, DEMONSTRATE that an updated catalog of earthquakes does not make the existing probabilistic SHA unviable.**
Table 4.3.10.1-4			
SHA-C1 through SHA-C5	----------	No objection	----------
Table 4.3.10.1-5			

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Index No.	Issue	Position	Resolution
SHA-D1 and SHA-D2	----------	No objection	----------
SHA-D3	The ground motion characterization model should include ground motion prediction equations (GMPEs) with alternative distance and magnitude scaling behaviors, not just a range of amplitudes.	Clarification	ENSURE that uncertainties are included in the model 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. …
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			

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

Index No.	Issue	Position	Resolution
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 andSFR-A2	----------	No objection	----------
Table 4.3.10.2-3			
SFR-B1	The differentiation between CC-I and CC-II is the use of "realistic" fragilities to improve plant representation and realism. Note S-N-17 negates the difference between the CCs and is therefore, inappropriate for CC-II.	Clarification	CC-II See Note ~~S-N-17,~~ S-N-20
SFR-B2 and SFR-B3	----------	No objection	----------
SFR-B4	The differentiation between CC-I and CC-II is the use of "realistic" fragilities to improve plant representation and realism. Note S-N-17 negates the difference between the CCs and is therefore, inappropriate for CC-II.	Clarification	CC-II See Note ~~S-N-17,~~ S-N-23
SFR-B5 and SFR-B6	----------	No objection	----------
Table 4.3.10.2-4			

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Index No.	Issue	Position	Resolution
SFR-C1	Justification of the selected basis needs to be provided, especially for cases where the basis in an extension or expansion of available information. Note S-N-27 also mentions "plant-specific justification" which is not reflected in the SR.	Clarification	SPECIFY the basis for screening of inherently rugged components **justifying the applicability to the plant and site or range of sites identified in SHA-A1.**
SFR-C2	Justification of the selected basis needs to be provided, especially for cases where the basis in an extension or expansion of available information. This comment is also supported by the discussion in Note S-N-28.	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 4.3.10.2-5			
SFR-D1	----------	No objection	----------
SFR-D2	A physical plant walkdown is extremely important in the development of a seismic PRA, including the fragilities of SSCs. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development. However, use of "investigations" in the SR without qualifiers can result in	Clarification	EVALUATE the seismic capacity of the as-designed, or as-built, or as-operated or as-intended-to-operate plant conditions via **a walkdown or, for PRAs performed during pre-operational phase,** investigation(s).

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Index No.	Issue	Position	Resolution
	unintended consequences.		
SFR-D3	----------	No objection	----------
SFR-D4	----------	No objection	----------
SFR-D5	The walkdown should also focus on operator pathways and potential unavailability of those pathways.	Clarification	EVALUATE potential functional and structural failure mechanisms, equipment anchorage, ~~and~~ support load path~~.~~**, and pathways necessary for performing required ex-control room actions.**
SFR-D6 and SFR-D7	----------	No objection	----------
SFR-D8	----------	No objection	----------
Table 4.3.10.2-6			
SFR-E1	The differentiation between CC-I and CC-II is the use of "realistic" fragilities to improve plant representation and realism. Note S-N-17 negates the difference between the CCs and is therefore, inappropriate for CC-II.	Clarification	CC-II See Note ~~S-N-17,~~ S-N-40
SFR-E2	Note S-N-9, which is referenced for CC-II of this SR is incorrect because this SR is for a specific site.	Clarification	CC-II See Note ~~S-N-9,~~ S-N-40
SFR-E3	Generic data needs to be shown to be applicable to an SSC because of the potential for SSCs in NLWR designs that may not correspond to established SSC categories and/or may	Qualification	CC-I 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

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Index No.	Issue	Position	Resolution
	not be in common SSC locations in the plant. This comment is applicable to both CC-I and CC-II. For pre-operational phase PRAs, the Standard supports use of a bounding site. As a result, estimating fragilities using generic or conservative assumptions needs to be shown to be bounding for the range of anticipated sites. Note S-N-42 identifies a similar intent. This comment is applicable only to CC-I. The differentiation between CC-I and CC-II is the use of "realistic" fragilities to improve plant representation and realism. Note S-N-17 negates the difference between the CCs and is therefore, inappropriate for CC-II.		qualification test data, and earthquake experience data) or conservative assumptions for the SSCs as being **applicable to the SSC and** appropriate for the plant or **applicable to the SSC and bounding for the range of sites identified in SHA-A1**. <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. See Note ~~**S-N-17,**~~ S-N-43
SFR-E4	Generic data needs to be shown to be applicable because of potential difference in the design and use of relays and similar devices in NLWR designs. This comment is applicable to both CC-I and CC-	Qualification	<u>CC-I</u> ESTIMATE contact chatter seismic fragilities for relays and other similar devices that affect SSCs identified in the Systems Analysis **justifying the use of generic fragility data or conservative assumptions as being applicable and appropriate for the plant or applicable and bounding for**

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Index No.	Issue	Position	Resolution
	II. For pre-operational phase PRAs, the Standard supports use of a bounding site. As a result, estimating fragilities using generic or conservative assumptions needs to be shown to be bounding for the range of anticipated sites. Note S-N-17 identifies a similar intent. This comment is applicable only to CC-I. The differentiation between CC-I and CC-II is the use of "realistic" fragilities to improve plant representation and realism. Note S-N-17 negates the difference between the CCs and is therefore, inappropriate for CC-II. Note S-N-45 pre-determines the outcome of design and SEL development. If relays or other similar devices are unimportant, they will not show up in the SEL. Therefore, note S-N-45 is unnecessary. This comment applies to both CC-I and CC-II.		**the range of sites identified in SHA-A1**. See Note S-N-17, S-N-44~~**, S-N-45**~~ <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. See Note ~~**S-N-17**~~, S-N-44~~**, S-N-45**~~

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Index No.	Issue	Position	Resolution
SFR-E5	Generic data needs to be shown to be applicable because of potential differences in the seismic-induced flood and fire sources in NLWR designs. This comment is applicable to both CC-I and CC-II. For pre-operational phase PRAs, the Standard supports use of a bounding site. As a result, estimating fragilities using generic or conservative assumptions needs to be shown to be bounding for the range of anticipated sites. Note S-N-17 identifies a similar intent. This comment is applicable only to CC-I. The differentiation between CC-I and CC-II is the use of "realistic" fragilities to improve plant representation and realism. Note S-N-17 negates the difference between the CCs and is therefore, inappropriate for CC-II.	Qualification	CC-I ESTIMATE seismic fragilities for credible seismic-induced flood sources (see Requirement SFR-D6) and seismic-induced fire ignition sources (see Requirement SFR-D7) **justifying the use of generic fragility data or conservative assumptions as being applicable and appropriate for the plant or applicable and bounding for the range of sites identified in SHA-A1**. See Note S-N-17 CC-II 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. ~~**See Note S-N-17**~~
SFR-E6	----------	No objection	----------
SFR-E7	This SR does not follow SFR-E6 in cross-referencing SPR-E8. SPR-E8 evaluates the impact	Qualification	For PRAs performed during the pre-operational stage, IDENTIFY assumptions made due to the lack of as-built, as-operated details that influence

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	of uncertainties and assumptions on the event sequence families, and consequently, the insights provided by the PRA. It appears that SFR-E7 only identifies and documents assumptions for pre-operational stage PRAs. It is recognized that the nature of the assumptions will be different for pre-operational stage PRAs compared to operational stage PRAs. However, this does not preclude the evaluation of their impact on the insights from the pre-operational phase PRA. The evaluation will also support preceding SRs where generic fragility values will potentially be chosen as "bounding" for a range of sites. See also the qualification for SPR-E8.		the Seismic Fragility Analysis **in a manner that supports Requirement SPR-E8**.
Table 4.3.10.2.7			
SFR-F1 and SFR-F2	----------	No objection	----------
SFR-F3	Cross-referencing SFR-E7 is desirable in the SR itself.	Clarification	…DOCUMENT assumptions and limitations due to the lack of as-built, as-operated details or site details associated with the Seismic Fragility Analysis **identified in SFR-E7. ~~See SFR-E7.~~**

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Index No.	Issue	Position	Resolution
Section 4.3.10.3			
Section text	----------	No objection	----------
Table 4.3.10.3-1			
HLR-SPR-A	----------	No objection	----------
HLR-SPR-B	The plant response model and event sequences should reflect the plant at the stage for which the PRA is developed or updated.	Clarification	The Seismic Plant Response Model shall include seismic-induced SSC failures, non-seismic-induced SSC failures, unavailabilities, human errors, POSs, sources of radioactive material, and multi-reactor effects that **represent the as-designed, or as-built, or as-operated or as-intended-to-operate and** can affect the frequencies of seismic-induced event sequence families modeled in the PRA.
HLR-SPR-C 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-B5	----------	No objection	----------
SPR-B6	The SR limits the inclusion of relays or similar devices to those that are "that are risk-significant contributors to frequencies of event sequence families modeled in the PRA." However, the risk-	Clarification	Using a systematic process, INCLUDE in the system analysis the effects of those relays or similar devices whose contact chatter results in the unavailability or spurious actuation of SSCs that are **included in the seismic equipment list developed to meet Requirement SPR-C1.** ~~risk-significant contributors to~~

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Index No.	Issue	Position	Resolution
	significance cannot be determined *a priori*.		~~frequencies of event sequence families modeled in the PRA.~~
SPR-B7 through SPR-B11	----------	No objection	----------
SPR-B12	The SR is worded incorrectly	Clarification	CC-I For all other secondary hazards explicitly retained in the seismic PRA, SATISFY the requirements of ~~**the following at Capability Category I**~~ Other External Hazards Fragility SRs of HLR-OFR-A and the Other External Hazards Plant Response Model SRs of HLR-OPR-B **at Capability Category I**, except where they are not applicable. CC-II For all other secondary hazards explicitly retained in the seismic PRA, SATISFY the requirements of ~~**the following at Capability Category II**~~ Other External Hazards Fragility SRs of HLR-OFR-A and the Other External Hazards Plant Response Model SRs of HLR-OPR-B **at Capability Category II**, except where they are not applicable.
SPR-B13	----------	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 must be assessed. Further, ESQ-C7 requires that human actions be feasible to use their	Qualification	CC-I and CC-II When addressing **feasibility,** influencing factors**,** and the timing considerations covered in Requirements **HR-G1,** HR-

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Index No.	Issue	Position	Resolution
	HEPs in event sequence quantification. The CC-II requirement incorrectly states that CC-I of HLR-HR-G should be satisfied.		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 influencing factors and the timing considerations covered in Requirements HR-G4, HR-G6, and HR-G8, INCLUDE the effect of the seismic hazard on the control room and ex-control room human actions.
Table 4.3.10.3-6			
SPR-E1 through SPR-E7	----------	No objection	----------
SPR-E8	This SR limits itself to PRAs other than the pre-operational stage through cross-references only to SFR-E6 and SPR-E6. It is recognized that the nature of the assumptions will be different for pre-operational stage PRAs compared to operational stage PRAs. However, this does not preclude the evaluation of their impact on the insights from the pre-operational phase PRA. In fact, due to the nature and extent of the assumptions made in the pre-	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 **and/or SFR-E7**, and system modeling, Requirement SPR-E6 **and/or SPR-E7**.

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Index No.	Issue	Position	Resolution
	operational phase PRA, an understanding of their impact on the insights and decisions using such a PRA is important.		
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	----------
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	For PRAs on plants prior to operation, the seismic fragility reviewers need to ensure that the assumptions made in the fragility development, including those for spatial considerations (e.g., locations, proximity etc.) are documented and consistent with the design.	Clarification	For PRAs on plants prior to operation… mechanisms. **The reviewers shall also ensure that the assumptions made in the fragility development, including those for spatial considerations (e.g., locations, proximity etc.) are documented and consistent with the design.**

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Index No.	Issue	Position	Resolution
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	----------
Nonmandatory Appendix S: Notes and Explanatory Material for Seismic PRA			
Section S.1			
Heading text	----------	No objection	----------
Table S-1			
S-N-1 through S-N-2	----------	No objection	----------
S-N-3	Note S-N-3 refers to RG 1.208, "A Performance-Based Approach to Define the Site-Specific Earthquake Ground Motion," as providing an acceptable approach to establishing a lower-bound magnitude for use in the hazard analysis. SHA-A6, consistent with state-of-practice, does not identify damage parameter CAV filter as a means of establishing the lower-bound magnitude. The NRC staff has also discouraged use of the damage parameter	Clarification	*Remove the following language in Note S-N-3*: ~~RG 1.208 [S-6] provides one acceptable approach to establishing a lower-bound magnitude for use in the hazard analysis.~~

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Index No.	Issue	Position	Resolution
	CAV filter in place of a lower bound magnitude for the PSHA. Use of CAV has often been misapplied in PSHAs to improperly filter out larger magnitude events at larger source-to-site distances. Recently completed PSHAs for NTTF Recommendation 2.1 and COL and ESP applications no longer use the CAV damage parameter in place of a lower bound magnitude. The NRC staff's related letter pursuant to 10 CFR 50.54(f) specified use of M5 (moment magnitude 5) as an appropriate lower-bound magnitude.		
S-N-4 through S-N-16	----------	No objection	----------
S-N-17	Generic data needs to be shown to be applicable because of potential differences in the seismic-induced flood and fire sources in NLWR designs. For pre-operational phase PRAs, the Standard supports use of a bounding site. As a result, estimating fragilities using generic or conservative	Clarification	… for the pre-operational stage PRA, it may be ~~**acceptable**~~ **necessary** to ~~**develop**~~ **use available** fragility estimates ~~**on some other basis**~~ (e.g., generic information), provided that the **estimates are justified to be applicable to the NLWR design and** the uncertainties introduced are acceptable for the application. For PRAs performed during the pre-operational stage, decoupling the fragility analysis from the hazard analysis to the

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Index No.	Issue	Position	Resolution
	assumptions needs to be shown to be bounding for the range of sites identified for the SPRA development.		extent practical allows for the fragility analysis to bounding a range of sites. It may be necessary to use generic data for this purpose **provided that the data is justified as being bounding for the range of sites identified for the SPRA development**.
S-N-18 through S-N-30	----------	No objection	----------
S-N-31	Generic data needs to be shown to be applicable because of potential differences in the seismic-induced flood and fire sources in NLWR designs. For pre-operational phase PRAs, the Standard supports use of a bounding site. As a result, estimating fragilities using generic or conservative assumptions needs to be shown to be bounding for the range of sites identified for the SPRA development.	Clarification	… for the pre-operational stage PRA, it may be ~~acceptable~~ **necessary** to ~~develop~~ **use available** fragility estimates ~~on some other basis~~ (e.g., generic information), provided that the **estimates are justified to be applicable to the NLWR design and** the uncertainties introduced are acceptable for the application. **For PRAs performed during the pre-operational stage, decoupling the fragility analysis from the hazard analysis to the extent practical allows for the fragility analysis to bounding a range of sites. It may be necessary to use generic data for this purpose provided that the data is justified as being bounding for the range of sites identified for the SPRA development.**
S-N-32	----------	No objection	----------
S-N-33	The note pre-supposes that there are no seismic vulnerabilities, which is inappropriate. Note S-N-35 discusses vulnerabilities. It is	Clarification	For PRAs performed during the pre-operational stage, ~~it is assumed that seismic vulnerabilities are absent and~~ the fragilities estimated based on design criteria or generic information ~~are~~ **may** be

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

Index No.	Issue	Position	Resolution
	highly likely that all vulnerabilities cannot be designed out because of other design constraints. Further, the design philosophy (e.g., design failure targets) may support the existence of vulnerabilities. Finally, at some seismic acceleration, vulnerabilities will exist. The use of generic data needs to be qualified as done in comments to Notes S-N-17 and S-N-31.		appropriate for PRA at the pre-operational stage provided that the **estimates are justified to be applicable to the NLWR design, the uncertainties introduced are acceptable for the application, and, if applicable, the data is justified as being bounding for the range of sites identified for the SPRA development**.
S-N-34 through S-N-41	----------	No objection	----------
S-N-42	Generic data needs to be shown to be applicable because of potential differences in the seismic-induced flood and fire sources in NLWR designs. For pre-operational phase PRAs, the Standard supports use of a bounding site. As a result, estimating fragilities using generic or conservative assumptions needs to be shown to be bounding for the range of sites identified for the SPRA development.	Clarification	… for the pre-operational stage PRA, it may be ~~**acceptable**~~ **necessary** to ~~**develop**~~ **use available** fragility estimates ~~**on some other basis**~~ (e.g., generic information), provided that the **estimates are justified to be applicable to the NLWR design and** the uncertainties introduced are acceptable for the application. For PRAs performed during the pre-operational stage, decoupling the fragility analysis from the hazard analysis to the extent practical allows for the fragility analysis to bounding a range of sites. It may be necessary to use generic data for this purpose **provided that the data is justified as being bounding for the range of sites identified for the SPRA**

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

Index No.	Issue	Position	Resolution
			development.
S-N-43 and S-N-44	----------	No objection	----------
S-N-45	Note S-N-45 pre-determines the outcome of design and SEL development. If relays or other similar devices are unimportant, they will not show up in the SEL. Therefore, note S-N-45 is unnecessary.	Clarification	**~~It is expected that relay chatter will not play an important role in the safety response for non-LWRs. As such, relays may not be included in the SEL making this SR not applicable.~~** **~~See SFR-E4~~**
S-N-46 through S-N-54	----------	No objection	----------
S-N-55	Note S-N-55 pre-determines the outcome of design and SEL development. If relays or other similar devices are unimportant or excluded, they will not show up in the SEL and consequently, in the SPRA	Clarification	**~~Some non-LWRs are expected to exclude relays from either their plant design safety case, thus meeting the intent of this SR by design.~~** **~~See SPR-B6~~**
S-N-56 and S-N-57	----------	No objection	----------

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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	The requirement does not address plant-specific hazards, which may not be identified as part of the identification of site-specific or design-specific hazards or hazard groups. Additionally, note HS-N-5 appears to be applicable to HS-A3 as it directly relates to plant-specific hazards and hazard groups.	Clarification	IDENTIFY site-**, plant-, or** ~~and~~ design-**specific** ~~unique~~ hazards and hazard groups**, as applicable to the stage of the plant lifecycle,** not already identified in Requirement HS-A2. See Notes HS-N3, HS-N-4, **HS-N-5**.
HS-A4	One of the NMA notes is assigned incorrectly, which may result in confusion when interpreting HS-A4. The relevant notes are the only notes that should be cross-referenced for a given supporting requirement.	Clarification	IDENTIFY secondary hazards associated with hazards and hazard groups from Requirements HS-A2 and HS-A3. See Notes HS-N-4, ~~HS-N-5,~~ HS-N-6.
Table 4.3.11.1-3			

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

Index No.	Issue	Position	Resolution
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 in the pre-operational stages of the plant lifecycle. As such, these types of reviews should be performed for any stage of the plant lifecycle.	Clarification	~~For PRAs performed on operating plants,~~ REVEW information about regional-, industrial-, governmental-, and plant-funded evaluations for each hazard, if available. See Note HS-N-6, HS-N-7
HS-B3 through HS-B4	----------	No objection	----------
HS-B5	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 requirement is effectively for qualitative screening, as per SCR-3 in Table 1.10-1 and	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); (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

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

Index No.	Issue	Position	Resolution
	because item (f) is a quantitative criterion, it should therefore not be included in the list.		response; ~~(f) the hazard or hazard group cannot produce a consequence above the value set in RI-A5.~~
HS-B6 through HS-B7	----------	No objection	----------
Table 4.3.11.1-4			
HS-C1 through HS-C6	----------	No objection	----------
HS-C7	Items (a) and (b) are missing words that complete the meaning of the statements.	Clarification	(a) for **a** discrete hazard, **taking** the product of... the hazard frequency and conditional failure probability of mitigating functions, as calculated in HS-C6; or (b) for **a** hazard characterized by...
HS-C8 through HS-C14	----------	No objection	----------
Table 4.3.11.1-5			
HS-D1	A physical plant walkdown is important for confirming assumptions used in hazard screening. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development. However, use of "investigations" in the SR without qualifiers can result in unintended consequences.	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 pre-operational phases,** as applicable, conditions through data and findings of investigation(s) of the plant (and its surroundings, as applicable to the hazard). See Note HS-N-18, HS-N-19.
Table 4.3.11.1-6			
HS-E1 through HS-E4	----------	No objection	----------
Section 4.3.11.2			

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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 text	----------	No objection	----------
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	----------
Section 4.3.11.3			
Section text	----------	No objection	----------
Nonmandatory Appendix HS: Notes and Explanatory Material for Hazards Screening Analysis			
Section HS.1			
Section text	----------	No objection	----------
Table HS-1			
HS-N-1 through HS-N-4	----------	No objection	----------
HS-N-5	The NMA note is assigned incorrectly, which may result in confusion when interpreting HS-A4. The relevant notes are the only notes that should be cross-referenced for a given supporting requirement.	Clarification	HS-A3 focuses on identifying plant-specific hazards that is **are** not inherent to the design or the site and can only be identified during or after plant construction (e.g., a decision to co-locate a chemical plant on the same site as the reactor). See ~~HS-A4~~ **HS-A3.**

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

Index No.	Issue	Position	Resolution
HS-N-6 through HS-N-12	----------	No objection	----------
HS-N-13	The statement includes extra words that obscure the meaning of the statement and may lead to misinterpretation.	Clarification	The term disposition means that peer review exceptions and deficiencies (e.g., Facts and Observations) ~~means a given exception or deficiency has~~ **have** either been resolved (i.e., closed) or has been shown to not impact the plant response model. See HS-C5.
HS-N-14	The note suggests that only CC-I would need to be met for system modeling in a screening model that; however, a user may always opt to satisfy the higher capability category.	Clarification	...Thus, new system modeling would**, at a minimum,** need to meet Capability Category I (CC-I) for any screening model.
HS-N-15 through HS-N-20	----------	No objection	----------

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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-WH-G	----------	No objection	----------
Table 4.3.12.1-2			
WHA-A1 through WHA-A4	----------	No objection	----------
WHA-A5	150 mile distance is arbitrary	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-A-6 through WHA-A-8	----------	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 4.3.12.1-6			

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

Index No.	Issue	Position	Resolution
WHA-E1 though WHA-E6	----------	No objection	----------
Table 4.3.12.1-7			
WHA-F1 though WHA-F4	----------	No objection	----------
Table 4.3.12.1-8			
WHA-G1 though 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	Include structures in the plant response analysis that may not be in the base plant model, if they protect other SSCs	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**
WFR-A2 through WFR-A8	----------	No objection	----------
WFR-A9	Clarify that must first develop correlations prior to assessing them	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**
Table 4.3.12.2-3			
WFR-B1 through WFR-B7	----------	No objection	----------
Table 4.3.12.2-4			

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

Index No.	Issue	Position	Resolution
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	Add item (f) for additional documentation requirement	Qualification	***f*.) the method of identifying SSC failure mechanisms, the identified failure mechanisms and associated failure modes. the treatment of wind pressure and APC effects, wind - generated missile effects, structural interactions effects and wind - driven rain effects if relevant to the plant, walkdown observations and conclusions and the results of the fragility evaluation."**
WFR-I2 and WFR-I3	----------	No objection	----------
Section 4.3.12.3			
Section text	----------	No objection	----------
Table 4.3.12.3-1			

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

Index No.	Issue	Position	Resolution
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	Phrase "additional, exclusive recovery actions" is confusing	Clarification	For treatment of additional, exclusive **operator** recovery actions relevant to the Wind Plant Response Analysis, SATISFY SRs of HLR HR-H, except where the requirements are not applicable.
WPR-D8 through WPR-D10	----------	No objection	----------
WPR-D11	The feasibility of operator actions in high winds PRA must be assessed. Further, ESQ-C7 requires that human actions be feasible to use their HEPs in event sequence quantification.	Qualification	<u>CC-I and CC-II</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.
Table 4.3.12.3-6			
WPR-E1 through WPR-E7	----------	No objection	----------
Table 4.3.12.3-7			

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

Index No.	Issue	Position	Resolution
WPR-F1 through WPR-F3	----------	No objection	----------
Section 4.3.12.4			
Section 4.3.12.4.1			
Section text	----------	No objection	----------
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	----------
Nonmandatory Appendix W: Notes and Explanatory Material for High Winds PRA			
Section W.1			
Section text	----------	No objection	----------
Table W-1			
W-N-1 through W-N-102	----------	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	This standard uses a term of investigation in SRs, replacing walkdown. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development.	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 **walkdown(s) or, for PRAs performed during pre-operational phase,** investigations.
XFHA-A9	----------	No objection	----------
Table 4.3.13.1-3			
XFHA-B1	----------	No objection	----------
XFHA-B2	XF-N-19 indicates that XFHA-B2 is appliable to both operating plants and pre-operation stage. While XF-N-21 shows "This SR is not appliable to operating plants."	Clarification	See Note XF-N19, XF-N-20~~, XF-N-21~~

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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-B3 and XFHA-B4	----------	No objection	----------
Table 4.3.13.1-4			
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	This standard uses a term of investigation in SRs, replacing walkdown. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development.	Clarification	COLLECT information via **walkdown(s) or, for PRAs performed during pre-operational 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	----------

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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 4.3.13.2-1			
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			
XFFR-B1	This standard uses a term of investigation in SRs, replacing walkdown. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development.	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 fragility evaluation, such as establishing or confirming the location and characteristics of flood protection features, penetrations/seals, and drainage features.
XFFR-B2	This standard uses a term of investigation in SRs, replacing walkdown. It is recognized that such a walkdown may not be performed at certain stages of the design and PRA development.	Clarification	For operating reactors, ASSESS the condition (e.g., SSC degradation) and configuration of SSCs observed during the **walkdown(s) or, for PRAs performed during pre-operational phase,** investigation(s).
XFFR-B3 through XFFR-B5	----------	No objection	----------
Table 4.3.13.2-4			
XFFR-C1	XF-N-42 allows the pre-operational stage PRA using	Clarification	<u>CC-II:</u> …See Note ~~XF-N-42,~~ XF-N-56

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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 generic information for fragility estimate. The differentiation between CC-I and CC-II is the use of "realistic" fragilities to improve plant representation and realism. Note XF-N-42 negates the difference between the CCs and is therefore, inappropriate for CC-II.		
XFFR-C2	----------	No objection	----------
Table 4.3.13.2-5			
XFFR-D1	XF-N-42 allows the pre-operational stage PRA using the generic information for fragility estimate. The differentiation between CC-I and CC-II is the use of "realistic" fragilities to improve plant representation and realism. Note XF-N-42 negates the difference between the CCs and is therefore, inappropriate for CC-II.	Clarification	CC-II … See Note ~~XF-N-42,~~ XF-N-56
XFR-D2	XF-N-42 allows the pre-operational stage PRA using the generic information for	Clarification	CC-II … See Note ~~XF-N-42,~~ XF-N-57

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

Index No.	Issue	Position	Resolution
	fragility estimate. The differentiation between CC-I and CC-II is the use of "realistic" fragilities to improve plant representation and realism. Note XF-N-42 negates the difference between the CCs and is therefore, inappropriate for CC-II.		
XFFR-D3, and 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			

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

Index No.	Issue	Position	Resolution
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	The feasibility of operator actions in external flooding PRA must be assessed. Further, ESQ-C7 requires that human actions be feasible to use their HEPs in event sequence quantification. The CCI and CCII language should be consistent.	Qualification	CC-I When addressing **feasibility,** 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	----------
Section 4.3.13.4			

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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 4.3.13.4.1			
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 4.3.12.4.3.2			
Section 4.3.13.4.3.2.1			
Section text	----------	No objection	----------
Section 4.3.13.4.3.2.2			
Section text	----------	No objection	----------
Section 4.3.13.4.3.2.3			
Section text	Based on context of these sections, sections should be re-numbered.	Clarification	*Change section numbers to following*: ~~4.3.13.4.3.2.3~~**4.3.13.4.3.3**
Section 4.3.13.4.3.2.3.1			
Section text	Based on context of these sections, sections should be re-numbered.	Clarification	*Change section numbers to following:* ~~4.3.13.4.3.2.3.1~~**4.3.13.4.3.3.1**
Section 4.3.4.3.13.4.3.3.3.2			
Section text	Based on context of these sections, sections should be re-numbered.	Clarification	*Change section numbers to following,* ~~4.3.13.4.3.3.3.2~~**4.3.13.4.3.3.2**
Section 4.3.13.4.3.3.3.3			
Section text	Based on context of these sections, sections should be re-numbered.	Clarification	*Change section numbers to following*: ~~4.3.13.4.3.3.3.3~~**4.3.13.4.3.3.3**

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

Index No.	Issue	Position	Resolution
Nonmandatory Appendix XF: Notes and Explanatory Material for External Flooding PRA			
Section XF.1			
Section text	----------	No objection	----------
Table XF-1			
XF-N-1 through XF-N-15	----------	No objection	----------
XF-N-16	This note provides examples of combinations of mechanisms with items (a) and (b), which are included in XF-N-14. It is not clear the examples of combinations of mechanisms	Clarification	… ~~Examples of combinations of mechanisms include the following:~~ ~~*(a)* "relevant deficiencies" refers to deficiencies that are relevant in the context of the screening conclusions and may include degraded condition of flood diversion features, flood protection, barriers, or drainage systems;~~ ~~*(b)* the plant information noted in this SR may include documents such as corrective action program entries.~~
XF-N-17 through XF-N-59	----------	No objection	----------
XF-N-60	Use of the term "unscreened flood mechanism" suggests that such mechanisms may have been screened out and subsequently screened back into the PRA, as though the act of screening had been undone. However, this term is interpreted as meaning flood mechanisms that were included in the PRA for consideration and evaluation (i.e., screened in).	Clarification	This requirement is intended to identify relevant scenarios for each ~~unscreened~~ flood mechanism **included for consideration and evaluation in the PRA**....

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

Index No.	Issue	Position	Resolution
XF-N-61 through XF-N-84	----------	No objection	----------

Pre-decisional

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-A2	----------	No objection	----------
OHA-A3	One of the NMA notes is assigned incorrectly, which may result in confusion when interpreting OHA-A3. The relevant notes are the only notes that should be cross-referenced for a given supporting requirement. Note O-N-5 is not applicable to this SR.	Clarification	CC-I USE regional data or generic data with analyst judgment in the hazard analysis. ~~See Note O-N-5~~ CC-II USE plant-specific data for the hazard analysis. INCLUDE, as necessary, regional data, generic information, and/or analyst judgment. ~~See Note O-N-5~~
OHA-A4	One of the NMA notes is assigned incorrectly, which may result in confusion when interpreting OHA-A4. The relevant notes are the only notes that should be cross-referenced for a given supporting requirement. Note O-N-5 is applicable to this SR. O-N-6 is not applicable to this SR.	Clarification	CC-I When using generic data, DEMONSTRATE that use of generic data is applicable and conservative. See Note ~~O-N-6~~ **O-N-5** CC-II When using generic data, DEMONSTRATE that use of generic data is applicable. See Note ~~O-N-6~~ **O-N-5**

Pre-decisional

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

Index No.	Issue	Position	Resolution
OHA-A5	One of the NMA notes is assigned incorrectly, which may result in confusion when interpreting OHA-A5. The relevant notes are the only notes that should be cross-referenced for a given supporting requirement. Note O-N-6 is applicable to this SR. O-N-7 is not applicable to this SR.	Clarification	When generating hazard curves, USE a parameter that most accurately represents a measure of the intensity of the hazard. See Note ~~O-N-7~~ **O-N-6**
OHA-A6	One of the NMA notes is assigned incorrectly, which may result in confusion when interpreting OHA-A6. The relevant notes are the only notes that should be cross-referenced for a given supporting requirement. Note O-N-7 is applicable to this SR.	Clarification	CALCULATE a family of hazard curves and DERIVE a mean hazard curve accounting for model and parameter uncertainties. **See Note O-N-7**
OHA-A7 through OHA-A10	----------	No objection	----------
Table 4.3.14.1-3			
OHA-B1 though 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			

Pre-decisional

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

Index No.	Issue	Position	Resolution
OFR-A4 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-B through HLR-OPR-E	----------	No objection	----------
Table 4.3.14.3-2			
OPR-A1 through OPR-A2	----------	No objection	----------
OPR-A3	The existing SR seems to suggest that an initiating event included in the plant response analysis could potentially be removed if it is not reflected by industry experience (e.g., the event hasn't yet occurred).	Clarification	**ENSURE that** ~~INCLUDE consideration of~~ initiating events **reflected** by industry experience (e.g., through review of plant-specific response to past related hazard events, industry operating experience, and other available related hazard evaluations for nuclear plants) **are included in the other hazard plant response analysis.**
OPR-A4	It may not be known whether an event causes a risk-significant event sequence and/or risk-significant event progression sequences unless it is included in the plant response model.	Clarification	INCLUDE in the plant response model the events identified by Requirements OPR-A1, OPR-A2, OPR-A3 above that ~~cause risk-significant event sequences and/or risk-significant event progression sequences~~ **lead to radiological consequences.**
Table 4.3.14.3-3			
OPR-B1 through OPR-B12	----------	No objection	----------

Predecisional

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 4.3.14.3-4			
OPR-C1 through OPR-C5	----------	No objection	----------
OPR-C6	The feasibility of operator actions in other-hazards PRA must be assessed. Further, ESQ-C7 requires that human actions be feasible to use their HEPs in event sequence quantification.	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			
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			

Pre-decisional

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 4.3.14.4.3.1			
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	----------
Nonmandatory Appendix O: Notes and Explanatory Material for Other Hazard PRA			
Section O.1			
Section text	----------	No objection	----------
Table O-1			
O-N-1 through O-N-4	----------	No objection	----------
O-N-5	O-N-5 is not applicable to OHA-A2.	Clarification	… See ~~OHA-A3~~ **OHA-A4**
O-N-6	O-N-6 is not applicable to OHA-4.	Clarification	… ~~See OHA-A4~~ **OHA-A5**
O-N-7	O-N-7 is not applicable to OHA-5.	Clarification	… See ~~OHA-A5~~ **OHA-A6**
O-N-8 through O-N-10	----------	No objection	----------
O-N-11	Sentences are not correct.	Clarification	Typical fragility analysis approaches ~~of~~ **for** structures and components may not be directly ~~apply~~ **applicable** to some hazards. The plant system response analysis may ~~be to~~ model the human action (i.e., type and timing) to prevent and mitigate event sequences. …

Pre-decisional

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

Index No.	Issue	Position	Resolution
O-N-12 through O-N-21	----------	No objection	----------

Pre-decisional

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			

Pre-decisional

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	----------
Section 4.3.15.3			
Section text	----------	No objection	----------
Nonmandatory Appendix ESQ: Notes and Explanatory Material for Event Sequence Quantification			
ESQ.1			
Heading text	----------	No objection	----------
Table ESQ-1			
ESQ-N1 trough ESQ-N-21	----------	No objection	----------

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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	----------
Nonmandatory Appendix MS: Notes and Explanatory Material for Mechanistic Source Term Analysis			

Pre-decisional

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 MS.1			
Heading text	----------	No objection	----------
Table ES-1			
MS-N-1 through MS-N-11	----------	No objection	----------

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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	Clarification	… 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; these quantities may be expressed in terms of inventories and release fractions** ~~the release fraction of each radionuclide group~~; …
RCRE-A3	----------	No objection	----------
Table 4.3.17.1-3			
RCRE-B1	----------	No objection	----------
RCRE-B2	----------	No objection	----------
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-RCRE-A through HLR-RCRE-C	----------	No objection	----------
Table 4.3.17.2-2			

Pre-decisional

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

Index No.	Issue	Position	Resolution
RCPA-A1 and RCPA-A2	----------	No objection	**----------**
RCPA-A3	Justification of input sources is primarily needed when multiple recognized sources exist but may recommend different protective action criteria	Clarification	… ~~JUSTIFY the use of these applicable documents (e.g., local requirements are more stringent than national requirements, use of international standards in lieu of U.S. standards).~~ **JUSTIFY the input sources used 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).**
RCPA-A4	The characteristics of the emergency response cohorts modeled are based on the requirements of the analysis, and therefore the examples given should not be should not be implied to be part of the requirement.	Clarification	CC-I USE one cohort ~~(e.g., for those not complying with protective actions)~~. CC-II USE two or more cohorts in the protective-action modeling ~~(e.g., one cohort for those not complying with protective actions and another cohort for those complying)~~.
RCPA-A5	The characteristics of the emergency response cohorts modeled are based on the requirements of the analysis, and therefore the examples given should not be should not be implied to be part of the requirement.	Clarification	IDENTIFY assumptions regarding compliance with protective actions ~~(e.g., a uniform percentage of the population is assumed to not evacuate)~~ based on generic data sources. See Note RC-N-2
RCPA-A6 through RCPA-A9	----------	No objection	**----------**
RCPA-A10	RCPA-A10 requires site-specific	Qualification	CC-I **ESTIMATE the evacuation speed based**

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

Index No.	Issue	Position	Resolution
	evacuation speeds at CC1 when a site may not have yet been selected.		**on generic data sources. JUSTIFY the speed used**. CC-II ESTIMATE the evacuation speed based on evacuation studies specific to the site. A constant average evacuation speed for applicable cohort(s) may be used. …
RCPA-A11 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	RCME-A1 requires site-specific meteorology at CC1 when a site may not have yet been selected. Either generic or bounding meteorological data may be used prior to site selection.	Qualification	CC-I **COMPILE meteorological data records from the site or bounding site covering a range of sites where the technology may be deployed. JUSTIFY that the data are spatially representative of the site (i.e., source) location and the region. See Note RC-6, RC-7** CC-II COMPILE meteorological data records

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

Index No.	Issue	Position	Resolution
			from the site. JUSTIFY that the data are spatially representative of the site (i.e., source) location and the region See Note RC-N-6, RC-N-7
RCME-A2	The minimum amount of meteorological data to be evaluated is not defined.	Clarification	CC-I SELECT hourly meteorological data for ~~a~~ representative **annual cycle(s)** ~~one-year period~~ from ~~a~~ **one or more** location**s** representative of the source and its surroundings. **JUSTIFY the amount of meteorological data evaluated if it is less than the minimum required in Regulatory Guide 1.23** CC-II EVALUATE hourly meteorological data for multiple years from the site location to select **a representative annual cycle(s)** ~~one-year period of data~~ that is representative of current conditions. **JUSTIFY the amount of meteorological data evaluated if it is less than the minimum required in Regulatory Guide 1.23**
RCME-A3	The substitution process for filling in missing data should involve techniques from recognized sources, which may include the use of data from regional sources such as government weather service stations. Also, the definition of the 90% data recovery rate is unclear.	Clarification	CC-I … SUBSTITUTE data to complete the data set using interpolation techniques or techniques **from EPA or industry guidance** ~~from regional recognized sources~~ (e.g., government weather service stations) where on-site meteorological data are not available. See Note RC-N-8 CC-II … For missing data, USE data from a different tower elevation~~,~~ or **from** co-located ~~tower~~ or **representative instruments** (if available), adjusted to

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

Index No.	Issue	Position	Resolution
			complete the database. SUBSTITUTE data to complete the data set using interpolation techniques, substitution techniques, or techniques **from EPA or industry guidance** ~~from regional recognized sources~~ (e.g., government weather service stations) where on-site meteorological data are not available. …
RCME-A4	The NRC preferred reference for evaluating meteorological data quality is Regulatory Guide 1.23.	Clarification	CC-I JUSTIFY that the accuracy of compiled meteorological data is sufficient for the desired application. **JUSTIFY inclusion of data that are not in compliance with Regulatory Guide 1.23 or its equivalent.** See Note RC-N-8 CC-II COMPILE meteorological data that has been collected under a qualified system of calibrations, maintenance activities, ~~and~~ instrument exposure**, etc**. See Note RC-N-8 **JUSTIFY inclusion of data that are not in compliance with Regulatory Guide 1.23 or its equivalent.**
RCME-A5 and RCME-A6	----------	No objection	----------
RCME-A7	The NRC preferred method for stability class determination is the delta-T method described in Regulatory Guide 1.23.	Clarification	CC-I **USE the delta-T method described in Regulatory Guide 1.23 for determining stability class. If data are unavailable to support use of this method,** USE a simplified stability classification approach. See Note RC-N-10 CC-II **USE the delta-T method described in Regulatory Guide 1.23 for determining stability class. If data are unavailable to support use of this method,** USE a stability classification method from

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

Index No.	Issue	Position	Resolution
			recognized sources. See Note RC-N-11
RCME-A8	The NRC preferred reference for evaluating meteorological data quality is Regulatory Guide 1.23.	Clarification	REVIEW meteorological data for its accuracy to determine adequacy of data recovery and its validity. **JUSTIFY inclusion of data that are not in compliance with Regulatory Guide 1.23 or its equivalent.**
RCME-A9 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			
HLR-RCAD-A through HLR-RCAD-F	----------	No objection	----------
Table 4.3.17.4-2			
RCAD-A1 through RCAD-A4	----------	No objection	----------
RCAD-A5	The requirement to include wind measurements that are reasonably representative of plume travel speed is unclear.	Clarification	USE a model that includes wind measurements that are reasonably representative **of the height of the release** ~~of plume travel speed and/or release height.~~
RCAD-A6 through RCAD-A8	----------	No objection	----------
Table 4.3.17.4-3			
RCAD-B1	----------	No objection	----------

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

Index No.	Issue	Position	Resolution
RCAD-B2	Use of the term "5th percentile dispersion factor" may be unclear.	Clarification	CC-I DETERMINE bounding meteorological conditions to be used in the analysis (e.g., **the χ/Q value that is exceeded in 5 percent of the total number of hours in the annual data set without regard to wind direction or the maximum — over all sectors — of the sector-specific χ/Q value that is exceeded in 0.5 percent of the total number of hours in the annual data set**~~5th percentile dispersion factor~~).
Table 4.3.17.4-4			
RCAD-C1	Aerodynamic wake effects may result in downwash of plumes emitted into the wake of buildings.	Clarification	USE dispersion algorithms that characterize atmospheric transport and dispersion ~~from elevated~~ **representative of** release heights~~, such as the tops of buildings or stacks~~. **INCLUDE downwash effects that may lower the effective plume height when using dispersion algorithms that characterize atmospheric transport and dispersion from representative release heights**.
RCAD-C2 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 and RCAD-E2	----------	No objection	----------
RCAD-E3	This supporting requirement is intended to address depletion of the plume by either wet or dry deposition, but the SR for CC2 does not explicitly	Clarification	CC-II ~~INCLUDE in the model either:~~ ~~(a) dry deposition and JUSTIFY the exclusion of wet deposition or~~ ~~(b) both dry and wet deposition~~. **INCLUDE dry and wet deposition source depletion**.

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

Index No.	Issue	Position	Resolution
	address the effect of deposition on depletion.		
RCAD-E4 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 some isotopes (e.g., tritium).	Clarification	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.
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 some isotopes (e.g., tritium).	Qualification	… JUSTIFY excluding any of the following pathways: (a) cloudshine; (b) groundshine; (c) skin deposition; (d) inhalation; (e) ingestion. **(f) skin absorption** See Note RC-N-16
RCDO-A2 through RCDO-A5	----------	No objection	----------

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

Index No.	Issue	Position	Resolution
RCDO-A6	The list of exposure pathways to be used in the analysis is be established by RCDO-A1.	Clarification	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 consistent with the results of RCDO-A1.**
RCDO-A7	----------	No objection	----------
RCDO-A8	The list of exposure pathways to be used in the analysis is be established by RCDO-A1.	Clarification	CC-I ~~DO NOT INCLUDE ingestion pathways in the model.~~ CC-II ~~USE generic intake quantities of foodstuffs and water.~~ CC-I and CC-II **MODEL ingestion pathways consistent with the results of RCDO-A1.** USE generic intake quantities of foodstuffs and water.
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			

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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-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	----------
Section4.3.17.7			
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	----------	No objection	----------
RCEC-B2	There is no requirement to justify use of generic data for cost parameter values at CC-II.	Qualification	CC-I ESTIMATE cost parameter values using regional data applicable to the site and generic data (as needed). CC-II **ESTIMATE cost parameter values using regional data applicable to the site and generic data (as needed).** **JUSTIFY use of generic data.**

Pre-decisional

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

Index No.	Issue	Position	Resolution
RCEC-B3	The supporting requirement implies that there is a choice between using recognized sources of cost data or using generic data. However, the counterpart to use of generic data is the use of regional data applicable to the site (see RCEC-B2). Use of either regional data applicable to the site or generic data should be based on use of recognized sources of cost data.	Qualification	USE recognized sources of cost data ~~or JUSTIFY the use of generic data~~. See Note RC-N-22
RCEC-B4 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 and RCQ-A2	----------	No objection	----------
RCQ-A3	Although the standard requires the analyst to compile a list of event	Clarification	COMPILE list of event sequence families and associated radiological consequences **in a manner that is consistent with the release category definitions in ES-C1**

Pre-decisional

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

Index No.	Issue	Position	Resolution
	sequence families and associated radiological consequences, there is no reference to the supporting requirements that provide that information (e.g., RCRE-A3, ES-C1, or MS-A).		**and the mechanistic source term parameters in HLR-MS-A**.
Table 4.3.17.8-3			
RCQ-B1 and RCQ-B2	----------	No objection	----------
RCQ-B3	It is unclear how the SRs of HLR-RI-B would be used to IDENTIFY risk-significant contributors. It is assumed that RI-B6 is intended. If so, RCQ-B3 seems redundant to RI-B6.	Clarification	~~IDENTIFY risk-significant contributors (elements) using the SRs of HLR-RI-B.~~ **IDENTIFY significant contributors to results of interest.** See Note RC-N-24
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	----------
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	----------

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

Index No.	Issue	Position	Resolution
RC.1 Nonmandatory Appendix RC			
RC-N-1 through RC-N-3	----------	No objection	**----------**
RC-N-4	Note RC-N-4 defines "local" as "“Local” refers to the geographical area associated with the plume exposure pathway emergency planning zone (e.g., approximately 10-mile radius)." However, it is conceivable that some NLWR applications may have an emergency planning zone of less than ten miles.	Clarification	“Local” refers to the geographical area ~~associated with the plume exposure pathway emergency planning zone (e.g., approximately 10-mile radius)~~ **within a radius of approximately ten miles**. “Regional” refers to the geographical area evaluated in the model that is beyond the local area (e.g., 10- to 50-mile radius). See RCPA-B1
RC-N5 through RC-N-7	----------	No objection	----------
RC-N-8	The application of the 90 percent data recovery rate criterion is unclear.	Clarification	**Data recovery is typically evaluated on an annual basis. The 90-percent rate applies to the composite of all variables (e.g., the joint frequency distribution of wind speed, wind direction, stability class) needed to model atmospheric dispersion for each potential release pathway. In addition, the 90-percent rate applies individually to the other meteorological parameters.** ANSI/ANS-3.11-2015 (R2020) [RC-6] provides information on qualified meteorologists and data substitution. Additionally, Table 1 of ANSI/ANS-3.11-2015 (R2020) establishes accuracies for each parameter. See RCME-A3, RCME-A4
RC-N9 through RC-N-11	----------	No objection	----------

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

Index No.	Issue	Position	Resolution
RC-N-12	Multiple commonly used variations of the Briggs plume rise algorithms exist.	Clarification	~~Acceptable~~ **Examples of** plume rise algorithms can be found in Briggs 1975 [RC-26]. See RCAD-C2
RC-N-13	Multiple commonly used building wake effect algorithms exist.	Clarification	~~Acceptable~~ **Examples of** algorithms that account for building wake effects are found in Slade 1968 [RC-22], Randerson 1984 [RC-23], Regulatory Guide 1.145 [RC-24], Regulatory Guide 1.194 [RC-25]. See RCAD-C3
RC-N-14 through RC-N-16	----------	No objection	----------
RC-N-17	Computation of a Total Effective Dose Equivalent (TEDE) as defined in 10 CFR 20.1003 requires the use of dose coefficients consistent with ICRP Publication 30.	Clarification	Examples of recognized sources for DCFs include: (a) ICRP (e.g., ICRP 60 [RC-13], ICRP 72 [RC-14]), and (b) Federal guidance reports (FGRs) (e.g., FGR-11 [RC-15], FGR-12 [RC-16], FGR-13 [RC-17]). **Computation of a Total Effective Dose Equivalent (TEDE) as defined in 10 CFR 20.1003 requires the use of dose coefficients consistent with ICRP Publication 30. Federal Guidance Report 11 provides tables of conversion factors in the column headed "effective" that are consistent with ICRP Publication 30. Likewise, the DDE is nominally equivalent to the effective dose equivalent (EDE) from external exposure if the whole body is irradiated uniformly. Since this is a reasonable assumption for submergence exposure situations, EDE may be used in lieu of DDE in determining the contribution of external dose from cloud submersion to the TEDE. Federal Guidance Report 12 provides dose coefficients in the column headed "effective".** See RCDO-B1
RC-N-18 through RC-N-24	----------	No objection	----------

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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	The staff do not consider reporting requirements when determining the acceptability of a PRA for a given application. Such reporting requirements should be provided by the appropriate regulatory authority on an application-specific basis.	Qualification	*This requirement does not need to be met to demonstrate PRA acceptability.*
RI-A5	The staff do not consider reporting requirements when determining the acceptability of a PRA for a given application. Such reporting requirements should be provided by the appropriate regulatory authority on an application-specific basis.	Qualification	*This requirement does not need to be met to demonstrate PRA acceptability.*

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

Index No.	Issue	Position	Resolution
Table 4.3.18.1-3			
RI-B1 through RI-B3	----------	No objection	----------
RI-B4	The use of the term "unscreened sources of radioactive material" suggests that such sources may have been screened out of a PRA, but subsequently screened back in as though the act of screening had been undone. 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 **considered and analyzed in the PRA** ~~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	----------

Pre-decisional

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

Index No.	Issue	Position	Resolution
Section 4.3.18.2.3			
Section text	----------	No objection	----------
Nonmandatory Appendix MS: Notes and Explanatory Material for Risk Integration			
Section RI.1			
Heading text	----------	No objection	----------
Table RI-1			
RI-N-1	Proper use of relative and absolute risk significance criteria.	Clarification	*Add this text*: **The choice between using relative or absolute risk significance criteria to develop a PRA should consider issues such as, but not limited to the following:** • **The use of absolute risk significance criteria may yield a limited set of risk-significant items that is insufficient for developing risk insights or verifying the PRA model.** • **Importance measures traditionally used in LWR PRAs to identify relative risk significant items (e.g., FV and RAW) may be inaccurate or misleading when applied to noncoherent logic models (i.e., logic models that contain NOT logic).** • **A PRA that is developed using absolute risk significance criteria should be revised if relative risk significance criteria are used to support a subsequent application, and vice versa.** • **The use of risk significance criteria (relative or absolute) should address the entire set of risk metrics computed by the PRA.**
RI-N-2	RG 1.247 is not the appropriate vehicle for endorsing anything related to LMP demonstrations.	Clarification	**Examples of** ~~Acceptable~~ absolute risk significance approaches to event sequence family and SSC risk significance were demonstrated in the Licensing Modernization Project's demonstration reports.

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

Index No.	Issue	Position	Resolution
RI-N-3	The staff do not consider reporting requirements when determining the acceptability of a PRA for a given application, such reporting requirements should be provided by the appropriate regulatory authority on an application-specific basis.	Clarification	*The reporting requirement in RI-A4 does not need to be met to demonstrate PRA acceptability.*
RI-N-4	The staff do not consider reporting requirements when determining the acceptability of a PRA for a given application. Such reporting requirements should be provided by the appropriate regulatory authority on an application-specific basis.	Clarification	*The reporting requirement in RI-A5 does not need to be met to demonstrate PRA acceptability.*
RI-N-5	The cited documents are useful examples, but do not constitute a comprehensive list.	Clarification	**Examples of** ~~The following~~ documents **that** provide ~~a~~ **guidance** ~~**for**~~ U.S light water reactor (LWR) risk informed licensing applications … best practices beyond their intended audiences **are:** ~~See~~ [RI-6], [RI-7], [RI-8], [RI-9], [RI-10], and [RI-11]. See RI-B2, RI-B3, RI-C1, RI-C2, RI-C3, RI-C4
RI-N-6 through RI-N-9	----------	No objection	----------

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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 thru CC-A6	----------	No objection	----------
Table 5.8-3			
CC-B1 thru 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	It is important to still document the process and results used to evaluate changes on previously implemented risk-informed decisions.	Qualification	**(h) record of the process and results used to evaluate changes on previously implemented risk-informed decisions**
CC-E2	----------	No objection	----------
Nonmandatory Appendix CC: Notes and Explanatory Material for Configuration Control			
Section CC.1			
Heading text	----------	No objection	----------
Table CC-1			

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

Index No.	Issue	Position	Resolution
CC-N1 and CC-N2	----------	No objection	----------

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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 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	----------

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

Index No.	Issue	Position	Resolution
Section 7.1			
Section text	This section states that, for plants in pre-operational stages, the requirements in Section 7 apply for newly developed methods that are introduced, "...following the first peer review performed according to the requirements in Section 6." This seems to suggest that the treatment of newly developed methods is dependent on the timing of the first peer review, not based on the definition of a NDM. An NDM introduced at any point before or after the first peer review, as stated, should be subjected to a peer review to assess the technical adequacy of the NDM, based on the requirements in Section 7.	Qualification	For PRAs performed on plants in the pre-operational stage, these requirements apply for newly developed methods that are introduced **at any point during the development of the PRA used to support an application.** ~~following the first peer review performed according to the requirements in Section 6.~~
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			

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

Index No.	Issue	Position	Resolution
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	The documentation list item (e) does not explicitly address documentation of data manipulation that may have been performed in support of the NDM, which may be important information for assessing the acceptability of an NDM.	Clarification	*(e)* the sources of data, and the collection process **and data manipulation performed** in support of the Newly Developed Method~~s~~;
NM-F2	----------	No Objection	----------
Nonmandatory Appendix NM: Notes and Explanation Material for Newly Developed Methods			
Section NM.1			
Heading text	----------	No Objection	----------
Table NM-1			
NM-N-1	----------	No Objection	----------

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REFERENCES[9]

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, New York, ANS, La Grange Park, Illinois, February 2021.

2. NRC, RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Washington, DC.

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[9] 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 help ensure that significant contributors to plant risk are not inadvertently excluded from the PRA. A hazard is a category of similar challenges to plant design or operations that poses some risk to a facility. A hazard group is a set of similar hazards that are assessed in a PRA using common approaches, methods, and likelihood data for characterizing the effect on the plant. Hazards represent events or phenomena that are generally classified as either internal hazards or external hazards, based on the defined plant boundary in a PRA. Hazards categorized under the internal events, internal flood, internal fire, seismic, high wind, and external flood hazard groups are typically analyzed and modeled quantitatively using a PRA. However, there are a number of internal and external hazards whose risk to a facility can be assessed qualitatively, quantitatively, or both, but in a simplified manner and without the need for a detailed PRA model. Regulatory position C.1.3.11 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 to a plant and a potential plant response that are too complex to be represented by a simplified analysis and should be modeled in detail using a PRA. This latter type of hazard is commonly referred to as an "other hazard" and regulatory position C.1.3.14 provides additional guidance on the modeling of such hazards.

A list of hazards and their potential impacts that should be considered include, but may not be limited to, those items listed in Tables B-1 and B-2. Table B-1 provides a list of hazards consistent with the list of hazards provided in Table HS-2 of the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) advanced non-LWR PRA standard, ASME/ANS RA-S-1.4-2021 , "Probabilistic Risk Assessment Standard for Advanced Non- Light Water Reactor Nuclear Power Plants" (Ref.1). However, 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 1993 (Ref. 2), and earlier nuclear power plant PRA studies. This list of hazards has evolved and expanded over the past several decades based on insights and lessons learned from other PRA-related programs and applications such as licensees' responses to Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities—10 CFR 50.54(f)," dated June 28, 1991 (Ref. 3). Table D-2 provides a list of hazard causes and potential conditions to consider during the process of determining 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.

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Table B-1. List of Hazards

Hazard Group	Hazard	Direct or Secondary Impact of Hazard
Animals	*Animals*	Land-based or airborne animals that cause damage to plant equipment, such as loss of off-site 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, etc.) outside the plant boundary defined by the internal fire PRA.
	Non-Safety Building Fire	Direct (e.g., thermal effects) or indirect effects (e.g., generation of combustion products, propagation to safety-related 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 rain fall, 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.
	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.

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	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, etc.) 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.
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.

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	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 infiltrate 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 causes dynamic loading or freezing conditions as a result of ice coverage.
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 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, landslide, avalanche, dam failure, industrial accidents, landslide, seiche, tsunami, and vehicle accidents.

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	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, landslide, 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.
	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 structures, systems, or components (SSCs) at or in the plant's analyzed area causing functional failure. Secondary hazards resulting from an aircraft impact include, but are not necessarily limited to, fire.
	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 form on the ground or the surface of an earthbound object when the temperature of the ground or surface of the object falls below freezing. 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.

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	Transportation Vehicle Explosion	Offsite accidents involving transportation resulting in a release of hazardous materials or combustion products, an explosion, or a 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.
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.

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Table B-2. List of Hazard Causes and Conditions

Combustion/fire—resulting in burning, release of hot/toxic gases, release of combustion products, or heat causing functional failure of SSCs, the ability of the operator to perform, or both.
Debris effects—resulting in clogging of a liquid flow path or adversely affecting equipment performance.
Dynamic forces (from dynamic or static loading)—resulting in structural damage to SSCs causing functional failure of SSCs or impeded operator ability to perform actions, or both
Explosions—resulting in dynamic forces, fire, missiles, or gas releases
Effects on operator ability to do the following: • perform an action due to physical obstacles • perform a cognitive function • see • breathe • communicate • obtain available information (e.g., poor procedures, poor or no indication)
High energy arcs
Missiles—projectiles damaging structures and equipment
Physical obstruction—movement of structures or equipment reducing accessibility
Reduced air quality—combustion products or other airborne particulates affecting equipment or operator performance
Reduced availability of cooling water
Structural failure, including the following: • collapse • functional failure (e.g., break in containment, settlement of a structure) • loss of structural integrity
Thermal effects including the following: • heat transfer (radiative, conductive, or convective; advection (i.e., bulk transport of a fluid)) • steam
Water effects—water infiltration, submergence or spray causing corrosion, loss of electrical integrity (e.g., electrical short), clogging, inaccessibility, structural failure

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REFERENCES [10]

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, New York, ANS, La Grange Park, Illinois, February 2021.

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

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Pre-decisional

[10] 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.