U.S. NUCLEAR REGULATORY COMMISSION

REGULATORY GUIDE 1.253, Revision 0

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GUIDANCE FOR A TECHNOLOGY-INCLUSIVE CONTENT-OF-APPLICATION METHODOLOGY TO INFORM THE LICENSING BASIS AND CONTENT OF APPLICATIONS FOR LICENSES, CERTIFICATIONS, AND APPROVALS FOR NON-LIGHT-WATER REACTORS

A. INTRODUCTION

Purpose

This regulatory guide (RG) describes an approach that is acceptable to the staff of the U.S. Nuclear Regulatory Commission (NRC) for using a technology-inclusive content-of-application methodology to inform specific portions of the safety analysis report (SAR) included as part of a non-light-water reactor (non-LWR) license application. Specifically, this RG endorses the methodology described in Nuclear Energy Institute (NEI) report NEI 21-07, Revision 1, "Technology Inclusive Guidance for Non-Light Water Reactors, Safety Analysis Report Content: For Applicants Using the NEI 18-04 Methodology" issued February 2022 (Ref. 1), with clarifications and additions, where applicable, as one acceptable process for use in developing certain portions of the SAR for an application for a non-LWR construction permit (CP) or operating license (OL) under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 2), or for a combined license (COL), or design certification (DC) under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 3). As of the date of this RG, the NRC is developing a rule to amend Parts 50 and 52 (RIN 3150-Al66). The NRC staff notes this RG may need to be updated to conform to changes to Parts 50 and 52, if any, adopted through that rulemaking. Further, as of the date of this RG, the NRC is developing an optional performance-based, technology-inclusive regulatory framework for licensing nuclear power plants designated as 10 CFR Part 53, "Licensing and Regulation of Advanced Nuclear Reactors," (RIN 3150-AK31) and anticipates that this RG will be updated after promulgation of those regulations to address content of application considerations specific to the licensing processes in this framework.

Applicability

This RG applies to designers of non-LWRs and applicants for permits, licenses, and certifications under 10 CFR Part 50 and 10 CFR Part 52 for such reactors. Upon conclusion of the rulemaking

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underway to amend 10 CFR Parts 50 and 52, the NRC may update this RG, if necessary, to conform to changes to Parts 50 and 52 adopted through that rulemaking. The NRC staff envisions that the approach in this RG will also support the technology-inclusive, risk-informed, and performance-based licensing framework that is now under development and currently designated as 10 CFR Part 53 (RIN 3150-AK31). The NRC staff plans to update this RG to reflect these regulations after a final rule is promulgated to reflect any additional guidance unique to the content of applications under those regulations.

Applicable Regulations[1]

- 10 CFR Part 50 contains regulations for licensing production and utilization facilities.
  - 10 CFR 50.34, "Contents of applications; technical information," describes the minimum information required for (1) preliminary safety analysis reports supporting CP applications and (2) final safety analysis reports (FSARs) supporting OL applications.
- 10 CFR Part 52 governs the issuance of DCs, and COLs for nuclear power facilities.
  - 10 CFR 52.47, "Contents of applications; technical information," describes the information required to be included in FSARs supporting applications for standard DCs.
  - 10 CFR 52.79, "Contents of applications; technical information in final safety analysis report," describes the information required to be included in FSARs supporting applications for COLs.

Related Guidance

- "Policy Statement on the Regulation of Advanced Reactors" (Volume 73 of the *Federal Register* (FR), page 60612 (73 FR 60612); October 14, 2008) (Ref. 4), establishes the Commission's policy for advanced reactor designs to protect the environment and public health and safety and promote the common defense and security.
- RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 5), provides detailed guidance to the writers of SARs to allow for the standardization of information the NRC needs for reviewing CPs and OL applications.
- RG 1.181, "Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e)" (Ref. 6), provides methods the NRC staff finds acceptable for complying with the provisions of 10 CFR 50.71(e), which requires periodic development of updates to FSARs.
- RG 1.206, "Applications for Nuclear Power Plants" (Ref. 7), provides guidance on the format and content of applications for licenses, certifications, and approvals for nuclear power plants submitted to the NRC under 10 CFR Part 52. RG 1.206 specifies the information to be included in an application for a light-water reactor (LWR), although the guidance may also be generally useful for non-LWR applications.

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1 The staff notes that for advanced reactors, the NRC will determine the applicability of specific technical requirements in the regulations, or the need to define additional technical requirements based on the safety assessments for a particular design, on a case-by-case basis. Applicants seeking to use the NEI 18-04 approach in the development of an application for a standard design approval (SDA) or manufacturing license (ML) should engage in pre-application dialogue with the NRC.

- RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors” (Ref. 8), contains guidance on adapting the general design criteria in 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” to non-LWR designs. Non-LWR designers, applicants, and licensees may use this guidance to develop principal design criteria (PDC) for any non-LWR designs, as required by the applicable NRC regulations. RG 1.232 also contains guidance for modifying and supplementing the general design criteria to develop PDC for two types of non-LWR technologies: sodium-cooled fast reactors and modular high-temperature gas-cooled reactors.

- 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. 9), contains the NRC’s endorsement of the Licensing Modernization Project (LMP) methodology in NEI 18-04, Revision 1, “Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development,” (Ref. 10) for selecting licensing-basis events (LBEs); classifying structures, systems, and components (SSCs); and assessing the adequacy of defense in depth (DID). Non-LWR reactor designers, applicants, and licensees may use this guidance to develop the content of their applications for non-LWR designs. Specifically, for applicants following the guidance in NEI 21-07, Revision 1, the LMP methodology is the baseline approach for developing the application.

Purpose of Regulatory Guides

The NRC issues RGs to describe methods that are acceptable to the staff for implementing specific parts of the agency’s regulations, to explain techniques that the staff uses in evaluating specific issues or postulated events, and to describe information that the staff needs in its review of applications for permits and licenses. Regulatory guides are not NRC regulations and compliance with them is not required. Methods and solutions that differ from those set forth in RGs are acceptable if supported by a basis for the issuance or continuance of a permit or license by the Commission.

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), under control 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 and 3150-0151), Attn: Desk Officer of the Nuclear Regulatory Commission, 725 17th Street, NW, Washington, DC, 20503.

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 the NRC staff's guidance on using a technology-inclusive content-of-application methodology to develop specific portions of the SAR included as part of a non-LWR license application. Specifically, this RG endorses the methodology described in NEI 21-07, Revision 1,[2] as one acceptable method for use in developing certain portions of the SAR for an application for a non-LWR CP or OL under 10 CFR Part 50, or COL or DC under 10 CFR Part 52, with clarifications and additions described below. The technology-inclusive methodology in NEI 21-07, Revision 1, provides a common approach for the development of those portions of the SAR that reflect the outcomes and insights from the implementation of the Licensing Modernization Project (LMP) methodology as described in NEI 18-04, Revision 1, and endorsed by the NRC in Regulatory Guide 1.233. The applicant is also responsible for demonstrating compliance with all applicable regulations and may request exemptions, as appropriate, to establish the licensing basis for the design.[3]

Background

As the NRC prepares to review and regulate a new generation of non-LWRs, the staff has recognized both the need to establish a flexible regulatory framework and the benefits of such a framework. The NRC described its efforts to prepare for possible licensing of non-LWR technologies in "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness," issued December 2016 (Ref. 11). In "NRC Non-Light Water Reactor Near-Term Implementation Action Plans," issued July 2017 (Ref. 12), and "NRC Non-Light Water Reactor Mid-Term and Long-Term Implementation Action Plans," issued July 2017 (Ref. 13), the NRC staff identified specific activities the agency would conduct in the near-term, mid-term, and long-term timeframes. In addition, the Commission encouraged the use of a performance-based, technology-inclusive licensing framework for small modular reactors in SRM-COMGBJ-10-0004/COMGEA-10-0001, "Staff Requirements—COMGBJ-10-0004/COMGEA-10-0001—Use of Risk Insights to Enhance Safety Focus of Small Modular Reactor Reviews," dated August 31, 2010 (Ref. 14), and SRM-SECY-11-0024, "Staff Requirements—SECY-11-0024—Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews," dated May 11, 2011 (Ref. 15). The NRC staff believes that this approach is appropriate to apply to the guidance development for non-LWRs.

Efforts to establish a risk-informed, performance-based, technology-inclusive regulatory framework for non-LWRs included the development of several key guidance documents. These include guidance on adapting the general design criteria in 10 CFR Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," for developing principal design criteria for non-LWR designs documented in RG 1.232. In addition, these also include the LMP described in the industry-developed guidance NEI 18-04, Revision 1, issued August 2019. NEI 18-04, Revision 1, was endorsed by the NRC in RG 1.233. NEI 18-04, Revision 1, provides a systematic, risk-informed, and technology-inclusive process for developing key inputs for the content of applications to improve the understanding of the

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2 The NRC encourages an LWR applicant proposing to use the risk-informed, performance-based process described in NEI 21-07, Revision 1, to engage in pre-application discussions with the NRC to provide information to the staff on its intended implementation of the NEI 21-07, Revision 1 methodology for its design.

3 The staff has provided guidance on which regulations apply to non-LWRs in Appendix B of the advanced reactor content of application project (ARCAP) Roadmap interim staff guidance (ISG) document, DANU-ISG-2022-01, "Review of Risk Informed, Technology Inclusive Advanced Reactor Applications—Roadmap" (Ref. 16).

safety and risk significance of system designs and the relationship of system functions to overall facility safety, specifically for non-LWR designs.

A key element of this new and flexible regulatory framework is to standardize the development of non-LWR application content to promote uniformity among applicants, support staff review consistency and predictability, and provide a well-defined base for evaluating proposed changes in review scope and requirements. The development of an application for an NRC license, permit, certification, or approval is a major undertaking, in that an applicant must provide sufficient information to support the agency's safety findings. The information and level of detail needed will vary according to whether an application is for a CP, DC, OL, COL, or other action.

The NRC staff has had success with a standard content-of-application methodology for large LWRs. RG 1.70, issued in the 1970s, and RG 1.206, issued in 2007 and revised in 2018, reflect the NRC's efforts to standardize the format and content of LWR applications. Guidance documents such as these and numerous others on specific technical areas address the suggested scope and level of detail for those applications.[4]

To standardize the development of advanced reactor application content, the staff has focused on two projects:

- advanced reactor content of application project (ARCAP)
- technology-inclusive content of application project (TICAP)

ARCAP is an NRC-led activity intended to provide guidance for a complete non-LWR application under either 10 CFR Part 50 or 10 CFR Part 52, and eventually the technology-inclusive, performance-based licensing framework for which a rule is now being developed as 10 CFR Part 53. As a result, ARCAP is broad, encompassing several industry-led and NRC-led guidance development efforts that aim to promote consistency in developing applications. As described in the ARCAP Roadmap ISG, a complete non-LWR application should include, among other things, an SAR that includes proposed technical specifications, an emergency plan, and other information such as physical security plans.

TICAP is an industry led guidance activity focused on the scope and depth of information to include in the portions of an SAR that address the implementation of the LMP methodology described in NEI 18-04, Revision 1, and endorsed by the NRC in Regulatory Guide 1.233. By focusing on those aspects of the facility design most relevant to the risks posed by non-LWR technologies, including design features and human actions, the TICAP guidance will help applicants provide sufficient information on the design and programmatic controls, while obviating the need for excessive detail in less important areas. The specific portions of the SAR within the scope of TICAP are described below in more detail. The ARCAP guidance encompasses and supplements the TICAP guidance. In particular, the ARCAP documents address areas of the SAR that are outside the scope of the TICAP guidance (i.e., not covered by the LMP process), such as technical specifications, control of routine plant effluents, control of occupational exposure, etc.

4 In addition, NUREG 0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. 17), provides guidance to the staff on how to review applications.

As a result of extensive public discussion on TICAP and ARCAP with industry and other external stakeholders, the NRC has proposed a 12-chapter structure for the SAR for a non-LWR application. In contrast, the SAR for large LWRs, as described in RG 1.206, has 19 chapters. The staff on occasion adds guidance to its structure to discuss matters not evaluated in other review guidance chapters. The 12 chapters proposed for an advanced reactor SAR, consistent with ARCAP/TICAP guidance and the methodology in this RG, are as follows:

Chapter 1, "General Plant and Site Description, and Overview of the Safety Analysis"
Chapter 2, "Methodologies, Analyses, and Site Evaluations"[5]
Chapter 3, "Licensing-Basis Events"
Chapter 4, "Integrated Evaluations"
Chapter 5, "Safety Functions, Design Criteria, and Structure, System, and Component Safety Classifications"
Chapter 6, "Safety-Related (SR) Structure, System, and Component Criteria and Capabilities"
Chapter 7, "Non-Safety-Related with Special Treatment (NSRST) Structure, System, and Component Criteria and Capabilities"
Chapter 8, "Plant Programs"
Chapter 9, "Control of Routine Plant Radioactive Effluents, Plant Contamination, and Solid Waste"
Chapter 10, "Control of Occupational Dose"
Chapter 11, "Organization and Human-System Considerations"
Chapter 12, "Post-Construction Inspections, Testing, and Analysis Programs"
Other documents incorporated by reference into the SAR (e.g., emergency plan)

For applications that follow the SAR structure above, the scope of TICAP as described in NEI 21-07, Revision 1, includes the first eight of these chapters (i.e., those informed by the LMP process).[6] Figure 1 illustrates the nexus between ARCAP, TICAP, and other guidance for an advanced reactor application.[7]

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5 The LMP process does not specifically address site evaluations, therefore, NEI 21-07, Revision 1, does not address this aspect for an advanced reactor application. However, PRA may be used to inform the design considerations of external hazards associated with potential sites. Therefore, the discussion on SAR content and organization provided in this TICAP RG is from the perspective of the overall application and includes guidance for aspects considered outside of the LMP process. ARCAP Chapter 2, also includes such guidance on application content on site evaluations that should be included in Chapter 2 of the SAR.

6 The NRC highly encourages pre-application engagement from applicants that plan to use the methodology in NEI-21-07, Revision 1, but rely on a different SAR structure than the 12-chapter approach described in this RG and addressed in ARCAP. Similarly, applicants following the 12-chapter SAR structure but not using the LMP approach of NEI 21-07, Revision 1, should engage the NRC staff early to ensure the application contains all the information required by regulations and to optimize application reviews. The Commission's 2008 "Policy Statement on the Regulation of Advanced Reactors," highlights the importance of pre-application activities.

7 Requirements for the contents of a final safety analysis report (FSAR) are provided in 10 CFR 50.34(b) and include items such as proposed technical specifications and emergency plans, as well as other technical and programmatic contents listed therein. It should be noted that items such as technical specifications and emergency plans may be incorporated by reference in the FSAR but are controlled by change processes other than 10 CFR 50.59 for OLs. For example, changes to the technical specifications, which are part of the license, require a license amendment, and emergency plan changes are controlled by 10 CFR 50.54(q).

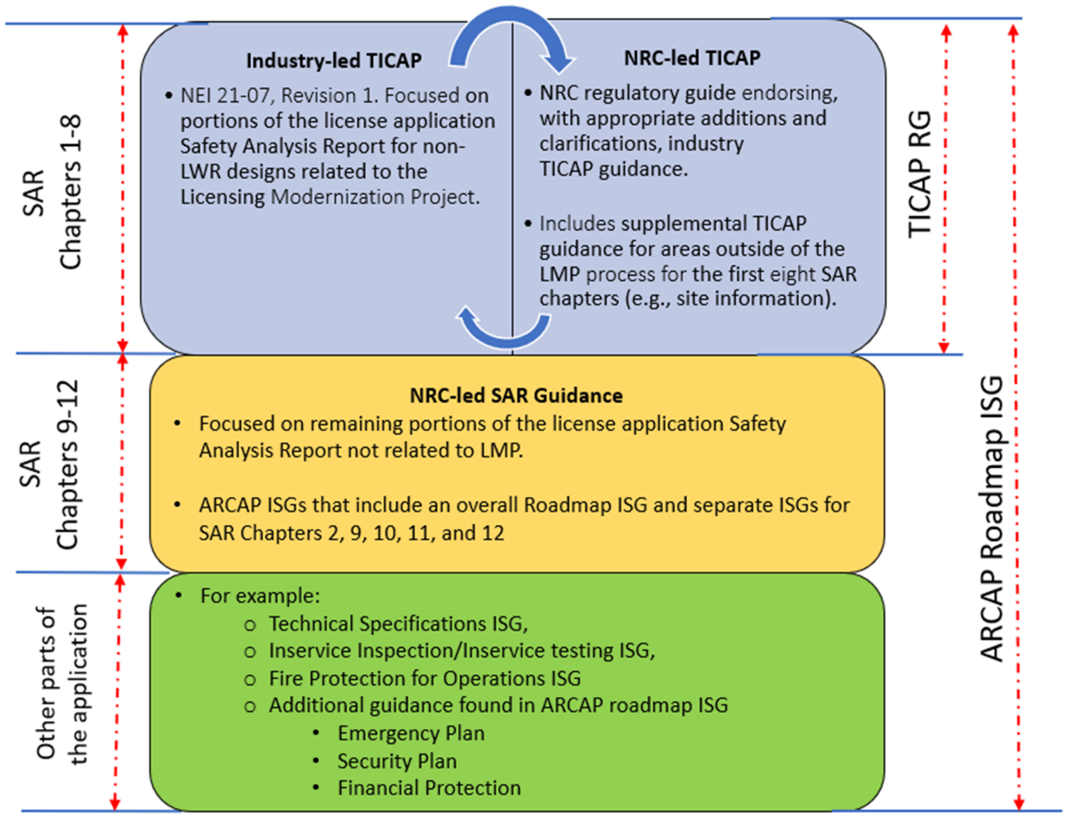

Figure 1. Relationship between ARCAP, TICAP, and the content of an application

Documents Endorsed in this Guide

After completing the TICAP efforts, NEI documented the results of the project as guidance in NEI 21-07, Revision 1 and submitted that guidance to the NRC for review and endorsement. The purpose of this RG is twofold:

1. To endorse NEI 21-07, Revision 1, with clarifications and additions. NEI 21-07, Revision 1, describes one acceptable approach for determining the scope and level of detail for the development of structured application content associated with the first eight chapters of the SAR. NEI 21-07, Revision 1, follows the LMP guidance and systematically describes the selection of LBEs; the classification and special treatment of SSCs; and the assessment of DID adequacy. Where applicable, this RG describes additional points of emphasis or further details relevant to the SAR sections discussed in NEI 21-07, Revision 1, and endorsed by this RG.

2. To provide additional guidance and information outside the scope of the LMP methodology and NEI 21-07, Revision 1, that the NRC staff has determined is also relevant and should be included as part of the application content related to the first eight chapters of the SAR.

Accordingly, this RG endorses NEI 21-07, Revision 1, with clarifications and additions, as one acceptable approach for use in developing certain portions of an SAR for a license, permit, or certification application to the NRC for a non-LWR using the methodology endorsed in RG 1.233. Additional details for each chapter appear in the corresponding section below.

In summary, the guidance in NEI 21-07, Revision 1, focuses on the portions of the SAR containing material addressed using the LMP process in NEI 18-04, Revision 1. The guidance in NEI 21-07, Revision 1, with the clarifications and additions described in this regulatory guide, will promote the submission of complete information to the NRC and ensure that application content is commensurate with the risk significance and complexity of the design and associated safety analysis. NEI 21-07, Revision 1, provides a standardized content development process to facilitate efficient SAR preparation by the applicant, NRC review of the application, and, if approved by the NRC, maintenance by the licensee. The guidance in NEI 21-07, Revision 1, should optimize the scope, content, and level of detail of each application, based on the risk significance and complexity of the design and associated safety analysis and the nexus between design elements and public health and safety.

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 (Ref. 18) and Management Directive and Handbook 6.6, "Regulatory Guides" (Ref. 19).

The following IAEA Safety Requirements were considered in the development of this Regulatory Guide:

- IAEA Specific Safety Requirements No. SSR-2/1, "Safety of Nuclear Power Plants: Design," issued 2016 (Ref. 20)

Documents Discussed in Staff Regulatory Guidance

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

C. STAFF REGULATORY GUIDANCE

This RG endorses the methodology described in NEI 21-07, Revision 1, as one acceptable method for use in developing certain portions of the SAR for an application for a non-LWR CP or OL under 10 CFR Part 50, or a COL or DC under 10 CFR Part 52. However, the NRC staff provides clarifications and additions to certain statements in NEI 21-07, Revision 1, as discussed below.

The guidance in this RG on the SAR scope, content, and level of detail is based on the appropriate level of design-specific information that should be provided in an application to the NRC to demonstrate that the facility design meets the regulatory standards for adequate protection of public health and safety. To provide effective and efficient technology-inclusive content guidance while ensuring the current application content requirements are met, this guidance describes an LMP-based safety analysis. The NRC highly encourages pre-application engagement between applicants and the staff to promote common understanding of proposed regulatory approaches, unique and novel designs, and technical issues, and to optimize resources and review schedules, especially for non-LMP-based applications.

The following sections describe the NRC's endorsement (with clarifications and additions, where applicable) of the corresponding chapters in NEI 21-07, Revision 1. In general, NEI 21-07, Revision 1, recommends that applicants first present the overall safety analysis for the reactor and then give supporting design and operational details in subsequent chapters. The staff notes that the methods, approaches, and data described in the regulatory guidance positions below are considered guidance and not requirements. However, in addition to presenting the overall safety analysis specific to their designs, applicants are required by the content of application requirements in Parts 50 and 52 to present a complete licensing basis by demonstrating compliance with applicable regulations, including any exemptions, where necessary, along with sufficient justification for each exemption. The suitability of such an exemption would be design-dependent, its justification would be the responsibility of the applicant, and the NRC would evaluate it on a case-specific basis.

NEI 21-07 explicitly addresses several licensing pathways: a combined license (COL) under 10 CFR Part 52 Subpart C; a design certification (DC) under 10 CFR Part 52 Subpart B; and a two-step license (CP/OL) under 10 CFR Part 50. An applicant using a licensing pathway other than one explicitly covered in NEI 21-07 may base the SAR content on the licensing pathway covered by NEI 21-07 and most similar to the approach it is using. For example, an applicant seeking a manufacturing license (ML) under 10 CFR Part 52 Subpart F or standard design approval (SDA) under 10 CFR Part 52 Subpart E may start with the guidance for a DC and make the necessary modifications to address its specific proposal and the applicable regulations. While such an approach reflects the similarity in the required contents of applications for the various licensing pathways in Parts 50 and 52, it will be up to the applicant to justify the guidance as applied and adjusted to address the scope of the application and to address the differences in the regulation for an ML or SDA application using the LMP methodology. As noted in Footnote 1, an ML or SDA applicant seeking to use RG 1.253 guidance should engage in a pre-application dialogue with the NRC.

1. Introduction and Development of Guidance

Section A, "Introduction," and Section B, "Development of Guidance," of NEI 21-07, Revision 1, discuss the document's purpose, background, scope, and organization, as well as the development of the guidance, an outline of the SAR, general instructions for use of the guidance, alternate licensing paths, two-step licensing (CP/OL), and DCs. Section C, "SAR Content Guidance," gives specific guidance on developing SAR content for a COL, with supplemental information for CP/OL and DC applications, where these differ from COL applications.

NEI 21-07, Revision 1, Section B.3, "General Instructions for Use of the Guidance," states the following:

> Italicized text provides background information for context and perspective. It is intended to provide readily accessible supporting information, but the italicized text does not require direct action on the part of the applicant. Information that is general in nature (e.g., general goals for level of detail, expectations for organization) will also be provided in italic.

C.1 Staff Position: NEI 21-07 Sections A and B provide acceptable background associated with TICAP guidance development with the following clarification:

a. The staff considers all discussion in NEI 21-07, Revision 1, to constitute guidance and not requirements; therefore, the staff considers the italicized text in NEI 21-07, Revision 1, to be part of the guidance and not simply background and context.

2. General Plant and Site Description and Overview of the Safety Analysis

Section C.1 of NEI 21-07, Revision 1, provides guidance for developing baseline information related to the plant description, the site description, the safety analysis based on the LMP methodology, and a summary of reference or source materials.

As described in NEI 21-07, Revision 1, the information in Chapter 1 of an SAR that follows NEI 21-07, Revision 1, should give the reviewer a basic understanding of the overall facility, such as the type of permit, license, certification, or approval requested; the number of reactor units; a brief description of the proposed plant location; and the type of advanced reactor being proposed. The site description should provide an overview of the actual physical, environmental, and demographic features of the proposed site, and how they relate to the safety analysis. For example, the site description should include geological, demographic, seismological, hydrological, and meteorological characteristics of the site and its vicinity.

In NEI 21-07, Revision 1, NEI defines the "affirmative safety case" as a collection of technical and programmatic information that demonstrates that the design meets the performance objectives of the technology-inclusive fundamental safety functions during design-specific anticipated operational occurrences (AOOs), design-basis events (DBEs), beyond-design-basis events (BDBEs), and design-basis accidents (DBAs). As described in NEI 21-07, Revision 1, section A.3., the "affirmative safety case" should do the following:

- Identify design-specific safety functions that are adequately performed by design-specific SSCs.
- Establish design-specific features to provide reasonable assurance that credited SSC functions are reliably performed and to demonstrate DID adequacy.

C.2 Staff Position: NEI 21-07, Revision 1, Section C.1, provides an acceptable method for developing baseline information related to the plant description, the site description, the overall safety analysis based on the LMP methodology, and a summary of reference or source materials with clarifications and additions as noted below.

a. Clarification: NEI 21-07, Revision 1, includes use of the terms "affirmative safety case," "safety case," and "licensing case." To avoid confusion and potential unforeseen consequences, applicants using NEI 21-07, Revision 1, should instead continue to use the established terminology in the current regulatory framework, including use of “safety analysis” and “licensing basis.”[8]

b. Clarification: The LMP methodology endorsed in RG 1.233 by its nature addresses off-normal conditions rather than normal operation. Applicants using NEI 21-07, Revision 1 to develop their SARs should also include additional information in parts of the SAR not derived from the LMP to describe and analyze normal operation. In addition, an applicant using NEI 21-07, Revision 1, is also responsible for demonstrating compliance with all applicable regulations, including exemptions, as necessary, with sufficient justification. Appendix B to the ARCAP Roadmap ISG contains staff guidance on which regulations apply to non-LWRs.

c. Addition: In addition to the information identified in NEI 21-07, Revision 1, Section C.1.1.2, on intended use of the reactor, applicants should also provide the nature (e.g., physical form) and inventory of contained radioactive materials in Chapter 1 or other appropriate sections of the SAR.

d. Addition: NEI 21-07 calls for the application to include discussions of the analyses of the potential radiological consequences from various event sequences identified from the PRA and related assessments. NEI 18-04 includes a specific question to be addressed during the integrated decision-making process related to the assessment of “cliff edge effects.” ASME/ANS RA-S-1.4-2021 addresses possible cliff-edge effects in areas such as seismic and flooding hazards. Any design requirements or special treatment of SSCs to prevent or mitigate cliff-edge effects should be included in the SSC specific descriptions in subsequent SAR chapters.

3. Methodologies, Analyses, and Site Evaluations

Section C.2 of NEI 21-07, Revision 1, presents guidance on the information to be included in the SAR on certain analyses and analytical tools (methodologies) used to identify LBEs, evaluate their consequences, or assess the performance of SSCs that are either safety-related (SR) or non-safety-related with special treatment (NSRST). The amount of information directly stated in this chapter, as opposed to incorporated by reference, could depend upon the extent of pre-licensing interactions between the applicant and the NRC, particularly interactions resulting in staff reviews and approvals (i.e., topical reports) and the extent to which the application relies on another license or certification (e.g., a COL referencing a certified design).

The information to be provided in Chapter 2 of an SAR following NEI 21-07, Revision 1, is primarily cross-cutting information or evaluations that support multiple LBEs or SSCs and provide a foundation for more specific information and analysis results given in other chapters of the SAR. The information provided in Chapter 2 should focus on the probabilistic risk assessment (PRA), source term analysis, DBA analytical methods, and other methodologies and analyses (e.g., civil and structural analysis, piping analysis, electrical load analysis, stress analysis, criticality analysis, thermal-hydraulic analysis, environmental qualification analysis, and dispersion modeling) that are pertinent to the LMP-based safety analysis.

When complete and final design information is not available at the CP application stage, the plant design and the associated PRA are considered preliminary, since they are less mature than they are at the OL stage. Therefore, the description of the PRA in a CP application should be a high-level overview or summary of topics such as the quality, scope, uses, and acceptability of the PRA. The applicant should provide justification that the PRA has been performed in such a way that the PRA results are reasonable

8 The NRC staff notes that neither the LMP methodology in NEI 18-04 nor the staff endorsement of that methodology in RG 1.233 use these terms, and no NRC regulation or guidance defines them. These terms are unnecessary to implement the LMP approach.

given the level of maturity of the design, and that the SAR provides sufficient information to support the CP findings. The applicant should also include any necessary commitments to upgrade and maintain the PRA so that its completion status at the OL stage is consistent with its intended uses. For a 10 CFR Part 52 application, the level of detail of the PRA in the application should be sufficient to meet the requirements in 10 CFR Part 52 that the SAR include a description of the PRA and its results.

The PRA is a model that provides an integrated assessment of the risk to the public from the nuclear power plant. The PRA identifies and assesses the sources of radionuclides in the plant and the various plant operating states which, for example, include full power, low power, and shutdown conditions for reactors. Chapter 2 of an SAR following NEI 21-07, Revision 1, describes the PRA at a summary level, addressing its scope, methodology, and pedigree (e.g., technical acceptability, peer review). RG 1.247 (for trial use), "Acceptability of Probabilistic Risk Assessment Results for Advanced Non-Light Water Reactor Risk-Informed Activities" (Ref. 21), endorses with exceptions and clarifications the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) 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. 22) and endorses with no exceptions and clarifications NEI 20-09, "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard" (Ref. 23).

RG 1.247 describes one approach acceptable to the NRC staff for determining whether a PRA used to support an application is sufficient to provide confidence in the results, such that the PRA can be used in regulatory decision making for non-LWRs. References may be included to other SAR chapters that discuss the use of the PRA and its results (e.g., selection of LBEs, evaluation of LBE risk significance against the LMP frequency-consequence targets (Figure 3-1 of NEI 18-04, Revision 1), determination of integrated risk and comparison to cumulative risk metrics, PRA uncertainties and assessment of DID adequacy, PRA safety functions, SSC safety classification, and reliability and capability targets).[9]

In Chapter 2 of an SAR following NEI 21-07, Revision 1, the applicant provides information on event sequence source terms specific to its design that is used in the LBE consequence analyses. The source term information should cover all radioactive material inventories and include the type, quantity, and timing of the release of radioactive material from the facility during LBEs. This chapter should include analysis methodologies, assumptions, bases, and justifications associated with transport of radioactive material from its point of origin to the accessible environment. For an LMP-based safety analysis, the application should include the use of a mechanistic source term, consistent with the advanced non-LWR PRA standard definition (see Appendix A to NEI 21-07, Revision 1). Mechanistic source term information that is common to some or all of the events considered for the plant may be given in Chapter 2 of an SAR following NEI 21-07, Revision 1, rather than repeated for each event. This information may include references to fuel qualification and performance topical reports and the associated NRC safety evaluations, as discussed in the supplementary information under Chapter 5.

C.3 Staff Position: Section C.2 of NEI 21-07, Revision 1, describes an acceptable method for developing baseline information related to the PRA (i.e., an overview of the PRA), source term analysis, DBA analytical methods, and other methodologies and analyses pertinent to the LMP-based safety analysis. Section C.2 of NEI 21-07, Revision 1 provides acceptable guidance on the discussion of the software and analytical tools used to perform the event sequence modeling and quantification, determine the mechanistic source terms, and perform radiological consequence evaluations for the LBEs and DBAs

9 The NRC encourages an LWR applicant that proposes to use NEI 21-07, Revision 1 to engage with the staff in pre-application discussions on its use of PRA tools and techniques during implementation of the LMP process and the development of SAR content for its design.

listed in Section C.3 of NEI 21-07, Revision 1 and the cumulative dose and risk calculations in Section C.4.1 of NEI 21-07, Revision 1. Section C.2 of NEI 21-07, Revision 1 also specifies that the applicant should identify the methods used, describe at a high level how they are applied to the radiological consequence evaluations, and describe the site characteristics modeled or site parameters postulated in the radiological consequence evaluations. The following clarifications and additions are included:

a. Clarification: NEI 21-07, Revision 1 calls for a "discussion of how the NRC RG that endorses the non-LWR PRA standard was implemented (pending finalization of the RG)." This regulatory guide has subsequently been issued as RG 1.247 for trial use.

b. Addition: In addition to the information that NEI 21-07, Revision 1, states applicants should include in SAR Chapter 2, the SAR Chapter 2 should discuss the analysis methods and assumptions for the total calculated radiological consequence dose at the EAB, the outer boundary of the low-population zone (LPZ), and the control room (if required, e.g., if operator actions are relied upon for safety-significant functions) to demonstrate that the facility meets the requirements of 10 CFR 50.34(a)(1)(ii)(D) or 10 CFR 52.79(a)(1)(vi) and the PDC for the control room (if applicable). Although an applicant is free to propose different approaches, two possible options for addressing these assessments based on the outcome of the LMP approach include:

   (1) Option 1: Use the DBA dose consequence results from an LMP-based approach to establish the acceptability of the EAB and LPZ. As described in RG 1.233, the DBA analysis under an LMP-based approach is a deterministic, conservative analysis that is analogous to the DBA analyses performed for new LWRs and operating reactors. Under this option, depending on the nature of the DBA, the application may need to include an exemption from the regulations in 10 CFR 50.34 or 10 CFR 52.79 that require an assumed "major accident"[10] to demonstrate containment performance and to confirm that the EAB and LPZ doses are below the reference values in the regulations. An applicant is responsible for justifying an alternative to using a major accident for this purpose.

   The uncertainty analyses for the mechanistic source terms and radiological doses should be described as part of the evaluation of conservative assumptions used in the DBA analysis. The plant design features intended to mitigate the radiological consequences of accidents, the site atmospheric dispersion characteristics, and the distances to the EAB and to the LPZ outer boundary are acceptable if the total calculated radiological consequences for the postulated fission product release meet the following reference values for public dose, given in 10 CFR 50.34(a)(1)(ii)(D) and 10 CFR 52.79(a)(1)(vi):

   - An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of the postulated fission product release would not receive a radiation dose in excess of 25 rem TEDE, and;

   - An individual located at any point on the outer boundary of the LPZ who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage) would not receive a radiation dose in excess of 25 rem TEDE.

10 For some non-LWR designs, assessment of the radiological consequences at the EAB and LPZ outer boundary using a major accident as specified in the regulations may be conservative or bounding and therefore establish compliance with 10 CFR 50.34 or 10 CFR 52.79. If assessment a major accident is not conservative or bounding for a specific non-LWR design in this regard or an applicant proposes to use a DBA that is not a major accident, justification for assessment of the radiological consequences at the EAB and LPZ outer boundary using a different event will be needed.

(2) Option 2: Use the greater of the dose consequence results from the bounding DBA and from a bounding BDBE, as identified in the LMP-based approach, to establish the acceptability of the EAB and LPZ. The uncertainty analyses for the mechanistic source terms and radiological doses should be described as part of the evaluation of conservative assumptions used in the analysis. This option provides an acceptable approach to compliance with 10 CFR 50.34 and 10 CFR 52.79 that precludes the need for an exemption from these requirements, as long as the bounding BDBE involves or bounds an event sequence meeting the description of a major accident and the offsite consequences are below the reference values for public dose in 10 CFR 50.34(a)(1)(ii)(D)(1) or 10 CFR 52.79(a)(1)(vi)(A) for the EAB and those in 10 CFR 50.34(a)(1)(ii)(D)(2) or 10 CFR 52.79(a)(1)(vi)(B) for the outer LPZ boundary.

c. Addition: Section C.2.1.1 of NEI 21-07, Revision 1, "Overview of PRA," includes a subsection titled, "Two-Step Licensing (CP Content)." This section notes that as part of a CP application the "applicant should address the last five items in the Section 2.1.1 list, consistent with the state of the plant design and the PRA at the time of the CP application." In addition to these five items, the application should include the item in the Section C.2.1.1 list labeled, "Identification of the sources of radionuclides addressed and the sources of radionuclides that were screened out."

d. Clarification: As noted in Section C.2.1.1 of NEI 21-07, a CP applicant should describe the attributes of the PRA in the application. In addition to these attributes, as amended by position C.3.c above, the CP application should also discuss topics such as the PRA's conformance to RG 1.247 for trial use, and NEI 20-09, if a peer review is performed at the CP stage. Appendix A of this RG 1.253, Revision 0, provides additional guidance on demonstrating the acceptability of the PRA supporting the CP application.

e. Clarification: In addition to the site information described in Section C.2 of NEI 21-07, Revision 1, the applicant in Chapter 2 of the SAR should provide information not developed using the LMP process, including summaries of the site-related information and analyses used to derive the design-basis hazard levels (DBHLs) documented in Chapter 6 of the SAR. The purpose of this information is (1) to demonstrate compliance with 10 CFR Part 100, "Reactor Site Criteria" (Ref. 24), Subpart B, "Evaluation Factors for Stationary Power Reactor Site Applications on or after January 10, 1997," and the relevant site-related requirements of 10 CFR Part 50 and 10 CFR Part 52, and (2) to describe the site characteristics used to inform the selection of the DBHLs in the design and safety analysis. Considerations for each relevant hazard include:

(1) NSRST SSCs credited in non-DBA licensing basis events (i.e., AOOs, DBEs, and BDBEs) or to establish adequate DID may need to be specially designed to withstand or be protected from the hazard (e.g., application of special treatments in accordance with NEI 18-04 and RG 1.233), and

(2) NSRST SSCs relied upon to establish adequate DID for beyond-design-basis hazards may need to be designed with special treatment to withstand or be protected from each such hazard.

The ARCAP ISG, DANU-ISG-2022-02, "Site Information," (Ref. 25), contains additional guidance on one acceptable approach to determining the scope and level of detail of the site information to be provided.

f. Addition: In addition to the information that NEI 21-07, Revision 1, states applicants should include in SAR Chapter 2, applicants should identify and describe the cross-cutting engineering analyses and methodologies used to establish their design bases or to confirm that intended safety functions will be fulfilled.

<u>4. Licensing Basis Events</u>

Section C.3 of NEI 21-07, Revision 1, provides guidance on the information related to the LBE selection methodology and the summary of LBEs (AOOs, DBEs, BDBEs, and DBAs) to include in an SAR. After identifying the LBEs, Chapter 3 of an SAR following NEI 21-07, Revision 1, should describe the systematic and reproducible process and methodology used to select the LBEs, and the specific analysis and evaluation of the selected LBEs for the proposed design. The analyses in this chapter are primarily described in terms of event sequences consisting of an initiating event, the plant response to the initiating event (which includes a sequence of successes and failures of mitigating systems), and a well-defined end state. Chapter 3 should also describe the process used to group and condense the many event sequences considered in the PRA into event sequence families that are used to define the AOOs, DBEs, and BDBEs. It is important to note that the term "event sequence" is used here, instead of the term "accident sequence" used in LWR PRA standards, because the scope of the LBEs also includes AOOs and initiating events that do not result in radioisotope releases.

It also important to note that for CP applicants, the requirements of 10 CFR 50.43(e)(1)(iii) to ensure that sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses do not apply. Accordingly, CP applicants are not required to provide evaluations of the safety margins using approved evaluation models. However, preliminary analyses should be available to demonstrate the following:

1. The design will provide sufficient safety margins during normal operations and transient conditions.

2. The applicant has identified the SSCs necessary to prevent accidents and mitigate accident consequences.

3. The applicant has demonstrated an understanding of the uncertainty associated with the performance of SSCs necessary to prevent accidents and mitigate accident consequences.

The items above are closely related; for example, an understanding of the uncertainties under item 3 is essential to an understanding of the margin under item 1. Additionally, items 2 and 3 support staff findings associated with 10 CFR 50.35(a)(3), namely, that the application describes the safety features and components that require research and development, and that the applicant will conduct a reasonably designed research and development program to resolve any associated safety questions (see the ARCAP Roadmap ISG on research and development).

C.4 Staff Position: NEI 21-07, Revision 1, Section C.3, on SAR Chapter 3, provides an acceptable method for developing information related to the LBE selection methodology and the summary of LBEs (AOOs, DBEs, BDBEs, and DBAs), with the following clarifications.

a. Addition and Clarification: The discussion of AOOs, DBEs, and BDBEs in Chapter 3 of the SAR should include a description of supporting data associated with the calculation of the mechanistic source terms and radiological consequences (to the extent that such information does not appear in the discussions of methodologies and analyses in Chapter 2, the descriptions of systems and functions in Chapters 5–7, or other sections of the SAR).

b. Addition: In addition to the material identified in NEI 21-07, Revision 1, Section C.3, that is derived by following the methodology of NEI 18-04, the applicant should address certain specified events and requirements in a new SAR Section 3.7, Special Event Analyses, as described below:

(1) 10 CFR 50.150(b) requires that the preliminary safety analysis report (PSAR) or FSAR include a description of (a) the design features and functional capabilities identified in 10 CFR 50.150(a)(1) (i.e., through the applicant's assessment required by section 50.150(a)(1)), and (b) how the design features and functional capabilities identified in 10 CFR 50.150(a)(1) meet the assessment requirements in 10 CFR 50.150(a)(1). The ARCAP Roadmap ISG contains guidance regarding aircraft impact assessments.

(2) Mitigation of Beyond-Design-Basis Events (MBDBE) (10 CFR 50.155, "Mitigation of beyond-design-basis events" (Ref. 26)): One of the primary lessons learned from the accident at the Fukushima Dai-ichi nuclear power plant in Japan was the significance of the challenge presented by a loss of multiple SR systems after a beyond-design-basis external event. As a result of lessons learned from the Fukushima Dai-ichi accident, the NRC amended its regulations to establish requirements for nuclear power reactor applicants and licensees for mitigating beyond-design-basis events (i.e., 10 CFR 50.155(b)(1)).

In the case of the Fukushima Dai-ichi accident, the loss of all alternating current power led to loss of core cooling, and ultimately to core damage and a loss of containment integrity. The design-basis for U.S. nuclear plants includes bounding analyses with margin for external events expected at each site. Extreme external events (e.g., seismic events or external flooding, etc.) beyond those accounted for in the design-basis, while unlikely, could present challenges to nuclear power plants. The following documents provide guidance on implementation of the regulations at 10 CFR 50.155 and applicants using NEI 21-07, Revision 1, to develop their applications should use the following documents:

- RG 1.226, "Flexible Mitigation Strategies for Beyond-Design-Basis Events" (Ref. 27), identifies methods and procedures the NRC staff considers acceptable for nuclear power reactor applicants and licensees to use to demonstrate compliance with NRC regulations on planning and preparedness for BDBEs as required by 10 CFR 50.155. RG 1.226 endorses, with clarifications, the methods and procedures in NEI 12-06, Revision 4, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," issued December 2016 (Ref. 28), as a process the NRC considers acceptable for meeting, in part, the regulations in 10 CFR 50.155. Additionally, RG 1.226 provides guidance for meeting the regulations in 10 CFR 50.155 in areas not covered by NEI 12-06, Revision 4.

- RG 1.227, "Wide-Range Spent Fuel Pool Level Instrumentation" (Ref. 29), identifies methods and procedures the NRC staff considers acceptable for demonstrating compliance with NRC regulations on providing a reliable means to remotely monitor wide-range spent fuel pool levels to support implementation of event mitigation and recovery actions as required by 10 CFR 50.155. RG 1.227 endorses, with exceptions and clarifications, the methods and procedures in NEI 12-02, Revision 1, "Industry Guidance for Compliance with NRC Order EA-12-051, 'To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation,'" issued August 2012 (Ref. 30), as a process the NRC staff considers acceptable for meeting certain regulations in 10 CFR 50.155.

- As noted in the statements of consideration for 10 CFR 50.155 (84 FR 39684) (Ref. 31), in recognition of the similarity of the existing extensive damage mitigation guidelines (EDMGs) formerly in 10 CFR 50.54(hh)(2) to the strategies required by 10 CFR 50.155(b)(1), the NRC relocated the EDMGs into the MBDBE rule as 10 CFR 50.155(b)(2). The EDMGs provide strategies and guidelines to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire. The EDMGs provide strategies and guidelines in the following areas: firefighting, operations to mitigate fuel damage, and actions to minimize radiological release. NEI 06-12, "B.5.b Phase 2 & 3 Submittal Guideline" (Ref. 32), provides guidance on how to develop the application content for demonstrating that the requirements of 10 CFR 50.155(b)(2) are met.[11]

Note: Applicants for a CP that are not requesting design finality for mitigation of specific beyond design basis events reflected in 10 CFR 50.155(a) and applicants for a DC are not required to provide information on this topic.

5. Integrated Evaluations

Section C.4 of NEI 21-07, Revision 1, provides guidance on documenting the integrated evaluations performed using the LMP process in NEI 18-04, Revision 1. Chapter 4 of an SAR following NEI 21-07, Revision 1, should provide the overall plant risk performance summary for the proposed design. This integrated plant evaluation assesses plant performance against the following three cumulative risk targets, and describes the margin between these targets and the predicted plant performance:

- The total mean frequency of exceeding a site boundary dose of 100 millirem from all LBEs should not exceed 1/plant-year. The value of 100 millirem is taken from the annual cumulative exposure limits in 10 CFR Part 20, "Standards for Protection against Radiation" (Ref. 33).

- The average individual risk of early fatality within 1 mile of the EAB from all LBEs, based on mean estimates of frequencies and consequences should not exceed 5×10^{-7}/plant-year, to meet the NRC safety goal QHO for early fatality risk.

- The average individual risk of latent cancer fatalities within 10 miles of the EAB from all LBEs, based on mean estimates of frequencies and consequences should not exceed 2×10^{-6}/plant-year, to meet the NRC safety goal QHO for latent cancer fatality risk.

Chapter 4 of an SAR following NEI 21-07, Revision 1, also documents the applicant's assessment of the adequacy of DID for the plant design, addressing the three focus areas for DID adequacy: plant capability; programmatic capability; and integrated risk-informed, performance-based DID adequacy. The baseline DID adequacy evaluation results in this chapter and other SAR chapters should be documented in sufficient detail so that, before being implemented, proposed future changes to physical, functional, operational, or programmatic features of the facility can be effectively evaluated for their potential to reduce DID.

C.5 Staff Position: NEI 21-07, Revision 1, Section C.4, on SAR Chapter 4, provides an acceptable method for developing information related to the integrated evaluations, which include the overall plant risk performance summary, margins between predicted plant performance and risk targets, and the documentation of DID adequacy with the following clarifications and additions:

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11 SRP Section 19.4, "Strategies and Guidance to Address Loss-of-Large Areas of the Plant Due to Explosions and Fires" provides guidance to the NRC staff for review of this topic. The SRP is intended to make information about regulatory matters widely available and to improve communication among the NRC, interested members of the public, and the nuclear power industry, thereby increasing understanding of the NRC's review process.

a. Clarification: The NRC anticipates that the DID discussion at the CP stage may be limited to plant capabilities because programmatic capabilities may not have been established yet. In addition, while not all plant capability DID attributes may be fully addressed at the CP stage, qualitative performance-based objectives for DID may be useful in establishing performance boundaries for final safety analysis report results. The CP application should provide a discussion in the SAR of the approach to establish DID adequacy. A discussion in the SAR to implement the DID adequacy assessment processes in RG 1.233 is considered acceptable for this purpose.

b. Addition: In addition to the results and margins, the SAR Chapter 4 should include a summary of departures taken from or unique inputs to the methodologies described in other chapters, if any, related to the analyses of cumulative risk measures. Examples that could arise due to factors such as the different time periods used in the assessments of licensing basis events and cumulative risk metrics could include sources of dose (cloud shine, inhalation, ground shine), additional inputs for dose conversion factors, and modeling assumptions (e.g., offsite protective actions). The summary can be provided via references to other documents or guidance related to the assessment of cumulative risk metrics.

c. Clarification: Human factors considerations for SSCs should be included in SAR Chapter 6 or 7, as appropriate. The human factors information in these SAR chapters should be consistent with the human factors information provided in SAR Chapter 11 in accordance with ARCAP DANU-ISG-2022-05, "Organization and Human Systems Considerations." (Ref. 34).

d. Clarification: Guidance for the change control process for the SAR, including ensuring the design and construction of defense-in-depth features remains adequate (i.e., up to issuance of an operating license), is addressed in NEI 18-04, Revision 1 as endorsed by RG 1.233. Additional guidance related to change control for the FSAR following issuance of the operating license is under development and the NRC is not taking a position on this topic at this time. The staff may address such change control processes in future regulatory actions, including possible rulemakings, license conditions, and development of guidance documents.

6. Safety Functions, Design Criteria, and SSC Safety Classifications

Section C.5 of NEI 21-07, Revision 1, provides guidance on the information related to safety functions, design criteria, and SSC classification established using the LMP process in NEI 18-04, Revision 1, and endorsed in RG 1.233. In the LMP process, LBEs are generally defined in terms of successes and failures of SSCs that perform safety functions and are modeled in the PRA. Therefore, the PRA safety functions (PSFs) are those functions credited for preventing or mitigating unplanned radiological releases from any source within the plant.

Chapter 5 of an SAR following NEI 21-07, Revision 1, should describe the applicant's approach to designating SSC safety functions and classifications in accordance with the PSFs. For SSCs, the applicant should describe the required safety functions (RSFs), the required functional design criteria (RFDC), the PDC, and the classification of SR and NSRST SSCs. These terms are defined in NEI 21-07, Revision 1, Appendix A, "Glossary of Terms."

C.6 Staff Position: NEI 21-07, Revision 1, Section C.5, provides an acceptable method for developing information related to the safety classification of SSCs, including information about RSFs, RFDC, PDC, and SR and NSRST SSCs with the following clarifications and additions:

a. NEI 21-07, Revision 1, Chapter 5, provides an acceptable approach for developing proposed PDC, with the following clarifications.

(1) Clarification: The inclusion of a proposed quality assurance PDC as described in Chapter 5 of NEI 21-07, Revision 1, is an acceptable method for implementing a graded approach to quality assurance for SSCs; it can also contribute to the basis for not addressing quality assurance in the scope of PDC in the more system- and component-specific PDC proposed.

(2) Clarification: As described in NEI 18-04, Revision 1, and RG 1.233, a non-LWR applicant may use a risk-informed methodology (e.g., the LMP methodology) to identify both RSFs and PSFs from which to determine RFDC and other special treatment requirements for SR and NSRST SSCs. The role of the RFDC and special treatment requirements derived from the LMP process in identifying design features and related attributes is similar to that of the advanced reactor design criteria and the requirements of the GDC. Therefore, to meet the regulations for proposing PDC, the scope of the proposed PDC should include SSCs important to safety. For applicants using the LMP process endorsed in RG 1.233, SSCs important to safety include both SR and NSRST SSCs. Therefore, the proposed PDC will need to address the functions provided by both SR and NSRST SSCs. NEI 21-07, Revision 1, Chapter 5, describes a two-tiered approach to PDC, comprising a higher-level portion based on meeting functional design criteria through RFDC and a bottom-up portion based on meeting specific performance requirements through complementary design criteria (CDC). This two-tiered approach proposed in NEI 21-07, Revision 1, divides PDC into PDC-RFDC and PDC-CDC.

(3) Clarification: Applicants adopting alternative approaches to proposing PDC based on similar risk-informed, performance-based licensing methodologies should provide suitable justification for their approaches and include any exemptions necessary. Exemptions from the regulations addressing content of applications are necessary if the full scope of PDC, as discussed above, is not addressed—that is, if the PDC do not cover all necessary design, fabrication, construction, testing, and performance requirements for all SSCs important to safety. For example, the justification may be that, to address specific elements of PDC scope not included here, the applicant has complied with other regulatory requirements that compel the applicant to provide the relevant information in other portions of the application.

b. Addition: Additional information on the role of fuel: In addition to the material identified in NEI 21-07, Revision 1, Section C.5, Chapter 5 of an SAR following NEI 21-07, Revision 1, should also address fuel qualification. The reactor core and its fuel are generally classified as SR because they are directly involved in performing fundamental safety functions. The application should provide the information for SR SSCs identified in NEI 21-07, Revision 1, Chapter 6, "Safety-Related SSC Criteria and Capabilities." However, the adequacy of fuel performance also depends on other information such as fuel design limits and fuel qualification, which the application should describe. The applicant's discussion should focus on the role of the fuel in the safety analysis for the reactor and on the adequacy of the plan to provide the basis for fuel performance as credited in the safety analysis. If not included elsewhere in the application or in referenced reports, this section of the SAR should include information sufficient to establish that:

(1) The role of the fuel in the safety analysis is adequately described. This can be accomplished by stating how the fuel will perform during (a) normal operation, including the effects of AOOs, and (b) accident conditions. To support these findings, sufficient information should be provided to clearly identify the design limits of the fuel and the fuel contribution in the accident source term. The applicant's discussion of the design limits and source term should address uncertainty from any limitations on data available, as reflected in the analyses discussed in Chapters 2 and 3 of NEI 21-07, Revision 1.

(2) The fuel qualification plan is adequate. The discussion of the fuel qualification plan should consider the proposed analysis methodologies (e.g., fuel performance codes), the use of existing data, and any ongoing testing or plans to use lead test specimens. If the applicant is using legacy data, it should justify the applicability of the data to the proposed facility (e.g., by confirming that the data were collected for a fuel fabricated consistent with the proposed fuel design and irradiated in an appropriate environment among other factors).

7. Safety-Related (SR) Systems, Structures, and Components Criteria and Capabilities

Section C.6 of NEI 21-07, Revision 1, provides guidance on the information related to the SR classification of SSCs, as well as their associated design criteria and performance capabilities. Chapter 6 of an SAR following NEI 21-07, Revision 1, should give details on SSCs classified as SR following the guidance in Chapter 5 of NEI 21-07, Revision 1. In particular, the SAR should give further detail on all design criteria and performance capabilities applying to SR SSCs, including safety-related design criteria, performance-based targets for reliability and capabilities, and special treatment requirements to provide sufficient confidence that the performance-based targets for the design will be achieved in the construction of the plant and maintained throughout the licensed plant life. For those SR SSCs whose reliability and capabilities have not been confirmed at the CP stage, the PSAR should include sufficient information (e.g., commitments for testing or research and development) to confirm that the reliability and capability performance targets informed by the final PRA will be met.

The term “special treatment” is derived from NRC regulations and NEI guidelines for implementing 10 CFR 50.69, “Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors.” RG 1.201, “Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance” (Ref. 35), defines special treatment as “those requirements that provide increased assurance beyond normal industrial practices that structures, systems, and components (SSCs) perform their design-basis functions.”

Special treatments are considered anything that is done, beyond procuring commercial-grade equipment, to provide increased assurance of the capability and reliability of both SR and NSRST SSCs, including, for example, design requirements, quality assurance requirements, availability controls, reliability and capability controls, and monitoring programs associated with reliability. Table 4-1 of NEI 18-04, Revision 1, gives additional information on possible types of special treatments that may be considered for an SSC. Chapter 6 of an SAR following NEI 21-07, Revision 1, should include information on the special treatments selected for SR SSCs.

One category of design requirements for SR SSCs consists of those measures or requirements needed to protect them from or ensure their ability to withstand the adverse effects of design-basis hazards when performing their RSFs. These design-basis hazards include both internal and external hazards; they are characterized as DBHLs that SR SSCs must have the ability to withstand or from which SR SSCs must be protected. DBHLs may be selected either deterministically or probabilistically. Chapter 6 of the SAR provides information on the establishment of the applicable DBHLs, the bases for establishment, and the associated parameters that lead to design requirements for SR SSCs.

C.7 Staff Position: NEI 21-07, Revision 1, Section C.6, on SAR Chapter 6, provides an acceptable method for developing information related to SSC design requirements and capabilities, including DBHLs, special treatment requirements, and system descriptions for SR SSCs with the following clarifications and additions:

a. Addition: In addition to describing the DBHLs as stated in NEI 21-07, Revision 1, Section C.6, the applicant may also use the guidance in section C.I.3 of RG 1.206, Revision 0 (Ref. 42),[12] to determine the information that should be included in Chapters 5 and 6 of the SAR regarding the translation of DBHLs to loads on SSCs, evaluation of those loads, and related design analysis. Pre-application interactions with the staff may be appropriate to determine the necessary level of information to be included in the SAR.

8. Non-Safety-Related with Special Treatment (NSRST) Structures, Systems, and Components Criteria and Capabilities

Section C.7 of NEI 21-07, Revision 1, provides guidance on the information related to the NSRST classification of SSCs, as well as their associated criteria and capabilities. Chapter 7 of an SAR following NEI 21-07, Revision 1, should describe the design and special treatment requirements for those SSCs classified as NSRST in Chapter 5 of the SAR. NSRST SSCs are not directly associated with RFDC (i.e., they are not SR SSCs) but are relied upon to perform risk-significant functions. Special treatments are defined above, with additional information provided in NEI 18-04, Revision 1. Risk-significant SSCs are those that perform functions that prevent any LBE from exceeding the frequency-consequence targets or that contribute significantly to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs. Appendix A to NEI 21-07, Revision 1, gives a more detailed definition of risk-significant SSCs. NSRST reliability and capability targets can be provided at the CP or the OL stage. For those NSRST SSCs whose reliability and capabilities have not been provided and confirmed at the CP stage, the application should include a discussion in the PSAR on how the applicant intends to confirm, at the OL stage, that reasonable reliability and capability performance targets have been established, align with the supporting analyses, and have special treatments defined to ensure the performance of SSCs meet the targets. The OL application should describe any testing and validation confirming NSRST SSC performance capabilities and availability, including any additional special treatments to be applied to the NSRST SSCs as compensatory measures to address a lack of operating experience.

C.8 Staff Position: NEI 21-07, Revision 1, Section C.7, provides an acceptable method for developing information related to the special treatment requirements for NSRST SSCs and the descriptions and capabilities of NSRST SSCs. Table 4-1 of NEI 18-04, Revision 1, gives additional information on the types of special treatments that may be considered for SSCs.

9. Plant Programs

Section C.8 of NEI 21-07, Revision 1, provides guidance on the information related to plant programs that support the LMP-based safety analysis. Chapter 8 of an SAR following NEI 21-07, Revision 1, should give an overview of the plant programs relied upon to support the LMP-based safety analysis, addressing these programs' purpose, scope, and performance objectives, as well as applicability to SR SSCs, NSRST SSCs, and operations activities. The applicant should describe the performance objectives of each program and explain how they relate to the targets or special treatments identified for SR and NSRST SSCs. This information should be included in the SAR or in documents that are incorporated by reference. Construction permit applications should include general descriptions in the SAR regarding any programs needed to implement special treatments and meet reliability and performance targets for SR SSCs and NSRST SSCs. These may include programs for inservice inspection/testing, maintenance, human factors, training, and reliability assurance.

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12 Chapter 3 of NUREG-0800 provides guidance to the NRC staff for review of this topic. The SRP is intended to make information about regulatory matters widely available and to improve communication among the NRC, interested members of the public, and the nuclear power industry, thereby increasing understanding of the NRC's review process.

Chapter 8 should cover those plant programs used for special treatments for SR and NSRST SSCs (as described in Chapters 6 and 7, respectively) to ensure that (1) reliability and performance targets are met, and (2) safety-significant uncertainties are addressed as part of DID. In addition, Chapter 8 should also identify and give an overview of the program or programs for documenting SSC reliability and capability targets, as described in Chapters 6 and 7 and ensuring that these targets are met. Program areas could also include human factors, quality assurance, startup testing, and equipment qualification, among others. The discussion of plant programs should address the different plant lifetime phases (i.e., design, construction, testing, and operations), as applicable.

C.9 **Staff Position**: NEI 21-07, Revision 1, Chapter 8, provides an acceptable method for developing information related to plant programs relied upon to support the LMP-based safety analyses, including programs used to implement special treatments for SR and NSRST SSCs and to meet reliability and capability targets.

D. IMPLEMENTATION

The NRC staff may use this regulatory guide as a reference in its regulatory processes, such as licensing, inspection, or enforcement. However, the NRC staff does not intend to use the guidance in this regulatory guide to support NRC staff actions in a manner that would constitute backfitting as that term is defined in 10 CFR 50.109, "Backfitting," and as described in NRC Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests" (Ref. 36), nor does the NRC staff intend to use the guidance to affect the issue finality of an approval under 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants." The staff also does not intend to use the guidance to support NRC staff actions in a manner that constitutes forward fitting as that term is defined and described in Management Directive 8.4. If a licensee believes that the NRC is using this RG in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfitting or forward fitting appeal with the NRC in accordance with the process in Management Directive 8.4.

ACRONYMS/ABBREVIATIONS

ACI	American Concrete Institute
ADAMS	Agencywide Documents Access and Management System
ANS	American Nuclear Society
AOO	anticipated operational occurrence
ARCAP	advanced reactor content of application project
ASME	American Society of Mechanical Engineers
BDBE	beyond-design-basis event
CDC	complementary design criterion/a
CFR	*Code of Federal Regulations*
COL	combined license
CP	construction permit
DC	design certification
DBA	design-basis accident
DBE	design-basis event
DBHL	design-basis hazard level
DG	draft regulatory guide
DID	defense in depth
EAB	exclusion area boundary
FSAR	final safety analysis report
IAEA	International Atomic Energy Agency
ISG	interim staff guidance
LBE	licensing-basis event
LMP	Licensing Modernization Project
LPZ	low-population zone
LWR	light-water reactor
ML	manufacturing license
NEI	Nuclear Energy Institute
NEIMA	Nuclear Energy Innovation and Modernization Act
NRC	U.S. Nuclear Regulatory Commission
NSRST	non-safety-related with special treatment
NST	non-safety-related with no special treatment
OL	operating license
OMB	Office of Management and Budget
PDC	principal design criterion/a
PRA	probabilistic risk assessment
PSAR	preliminary safety analysis report
PSF	PRA safety function
QHO	quantitative health objective
RFDC	required functional design criterion/a
RG	regulatory guide
RSF	required safety function
SAR	safety analysis report
SDA	standard design approval
SR	safety-related
SSC	structure, system, or component
TEDE	total effective dose equivalent
TICAP	technology-inclusive content of application project
U.S.C.	*United States Code*

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2. Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, "Domestic Licensing of Production and Utilization Facilities."

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

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

8. NRC, RG 1.232, "Guidance for Developing Principal Design Criteria for Non-Light-Water Reactors," Washington, DC.

9. 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, DC.

10. NEI 18-04, Revision 1, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light Water Reactor Licensing Basis Development," Washington DC, August 2019. (ML19241A472)

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13 Publicly available NRC published documents are available electronically through the NRC Library on the NRC's public website 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. For problems with ADAMS, contact the Public Document Room (PDR) staff at 301-415-4737 or (800) 397-4209, or email pdr.resource@nrc.gov. The NRC PDR, where you may also examine and order copies of publicly available documents, is open by appointment. To make an appointment to visit the PDR, please send an email to pdr.resource@nrc.gov or call 1-800-397-4209 or 301-415-4737, between 8 a.m. and 4 p.m. eastern time (ET), Monday through Friday, except Federal holidays.

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

11. NRC, "NRC Vision and Strategy: Safely Achieving Effective and Efficient Non-Light Water Reactor Mission Readiness," December 2016. (ADAMS Accession No. ML16356A670)

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Appendix A

Acceptability of a Probabilistic Risk Assessment That Supports a Non-Light-Water Reactor Construction Permit Application Based on the Licensing Modernization Project Methodology

A.1 Introduction

This appendix provides supplemental guidance on one approach that is acceptable to the U.S. Nuclear Regulatory Commission (NRC) staff for preparing a probabilistic risk assessment (PRA) for a non-light-water reactor (non-LWR) construction permit (CP) application (also referred to as a CP PRA[1]) under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. A-1), based on the Licensing Modernization Project (LMP) methodology in the Nuclear Energy Institute (NEI) report NEI 18-04, Revision 1, "Risk-Informed Performance-Based Guidance for Non-Light-Water Reactor Licensing Basis Development" (Ref. A-2). The NRC staff developed this guidance with the goal of informing all stakeholders, including applicants, of the type and detail of PRA information to be included in a non-LWR CP application that would be sufficient to provide confidence in the PRA results such that the PRA can be used in regulatory decision making. Accordingly, the application should demonstrate that:

- Commensurate with the preliminary plant design and proposed site described in the CP application, information developed from the PRA is sound and reliable.
- The PRA produces insights with appropriate fidelity to support implementation of the LMP methodology and development of the CP application.
- The CP applicant has defined processes and procedures to adequately maintain and upgrade the PRA to support continued implementation of the LMP methodology as the detailed plant design evolves and the plant is constructed, leading to submittal of the operating license (OL) application.

The term "PRA acceptability" describes the ability of a PRA to support risk-informed regulatory decision making and is defined in terms of meeting the NRC regulatory positions in Section C of Regulatory Guide (RG) 1.247 (for trial use), "Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities" (Ref. A-3). Specifically, Regulatory Position C.1 of RG 1.247 and its subsections provide guidance in the following four areas that are collectively assessed to determine the acceptability of a PRA:

1. <u>Scope of a PRA</u>: The scope of a PRA is defined in terms of (1) the metrics used to characterize risk, (2) the radiological sources that may contribute to risk, (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, and (4) the plant

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1 In this Appendix, a "CP PRA" refers to the collection of analyses represented by PRA documentation submitted as part of a CP application and the PRA documentation maintained by the applicant, which will be made available to the NRC staff through regulatory audits or in response to requests for additional information. Guidance on the four aspects of an acceptable CP PRA is provided in Sections A.3 through A.6 of this Appendix. Additionally, for the Technology-Inclusive Content of Application Project, the PRA includes the collection of analyses that represent risk contributors that are included in or screened out of the PRA logic model or that are accounted for by a risk-informed supplementary evaluation.

operating states (POSs) for which the risk is to be evaluated. The scope of a CP PRA is determined by its intended uses for representing the as-designed, as-to-be-built, and as-to-be-operated plant.

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

3. Elements of a PRA: The PRA elements are defined in terms of the fundamental technical analyses used to develop and quantify the CP 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.

4. Plant representation and PRA configuration control: Plant representation is defined in terms of how closely the CP PRA represents the plant as it is designed, built, and operated. In general, CP PRA results should be derived from a CP PRA model that represents the as-designed, as-to-be-built, and as-to-be-operated plant. Consequently, the CP PRA should be developed using an acceptable configuration control process.

The following sections provide guidance on these four interrelated areas of PRA acceptability and PRA documentation in the context of a CP application that is based on the LMP methodology. Consistent with NEI 18-04, Revision 1, and Section C.2.1.1 of NEI 21-07, Revision 1, "Technology-Inclusive Guidance for Non-Light Water Reactors, Safety Analysis Report Content: For Applicants Using the NEI 18-04 Methodology" (Ref. A-4), and their endorsements in 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. A-5), and the main body of this RG, this appendix assumes that the CP applicant will use the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) 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. A-6), to demonstrate the acceptability of the PRA. CP applicants may use other approaches to demonstrate the acceptability of the PRA with appropriate justification; the NRC staff will review such an approach on a case-by-case basis. An applicant should maintain more detailed PRA information that supports or confirms the acceptability of the PRA in plant records (i.e., archival documentation) that are available for audit by the NRC staff.

A.2 General

A.2.1 As discussed in Section B of Appendix A to DANU-ISG-2022-01, "Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications-Roadmap" (Ref. A-7), prospective CP applicants may find it beneficial to engage in pre-application activities with the NRC staff regarding approaches to demonstrating the acceptability of a CP PRA before the LMP-based CP application is submitted.

A.2.2 Consistent with NEI 21-07, Revision 1, Section 2.1.1, the CP applicant should clearly document in the PSAR the essential assumptions[2] made in developing the LMP-based safety analysis, which should include those assumptions relevant to the probability and consequence models, and the selection of elements to be incorporated in the CP PRA models.

A.2.3 The CP applicant should consider the near-term and long-term uses of the PRA as the PRA is developed to help ensure that it will be acceptable to support these uses. In addition to supporting implementation of the LMP methodology, results of the PRA may be used to demonstrate how certain regulations and Commission policies have been met and to support voluntary risk-informed applications, as discussed below.

Demonstrating that Certain Regulations Are Met

Currently, no regulation requires the development of a PRA to support a CP application under 10 CFR Part 50.[3] However, the CP applicant may use the PRA to demonstrate, in part, that the following regulations in 10 CFR Part 50 have been met:

1. 10 CFR 50.34(a)(1)(ii), which states, "It is expected that reactors will reflect through their design, construction and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products."

2. 10 CFR 50.34(a)(4), which requires the PSAR to include, "[a] preliminary analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility and including determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents."

Commission Policy Positions

The Commission's "Policy Statement on the Regulation of Advanced Reactors" (Volume 73 of the Federal Register (FR), page 60612 (73 FR 60612); October 14, 2008) (Ref. A-10) cites the following policy statements that express Commission expectations for use of the PRA. Specifically:

1. The advanced reactor policy statement articulates the expectation that advanced reactor designs will comply with the Commission's safety goal policy statement ("Safety Goals for Operations of Nuclear Power Plants; Policy Statement; Republication," 51 FR 28044; August 4, 1986, as corrected and republished at 51 FR 30028; August 21, 1986) (Ref. A-11). The safety goal policy

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2 The term *key assumption* is similar to the essential assumptions referred to in Section C.2.1.1 of NEI 21-07, Revision 1. A key assumption is defined in NUREG-2122 "Glossary of Risk-Related Terms in Support of Risk-Informed Decisionmaking" (Rev. A-8) as "[a]n assumption is considered to be key to a risk-informed decision when it could affect the PRA results that are being used in a decision and, consequently, may influence the decision being made."

3 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, includes proposed PRA-related requirements for 10 CFR Part 50 CP and OL applications that are similar to the existing PRA-related requirements for 10 CFR Part 52 licenses, certifications, and approvals. Further information about this rulemaking (including the proposed schedule) is available at https://www.nrc.gov/reading-rm/doc-collections/rulemaking-ruleforum/active/ruledetails.html?id=27.

statement broadly defines an acceptable level of radiological risk and establishes two qualitative safety goals which are supported by two quantitative objectives. Consistent with the safety goal policy statement, PRA is an acceptable tool for assessing conformance with the underlying purposes of the safety goals.

2. The advanced reactor policy statement notes that the Commission has issued a policy statement on "Severe Reactor Accidents Regarding Future Designs and Existing Plants" (50 FR 32138; August 8, 1985) (Ref. A-12), which indicates, in the context of the decision process for certifying a new standard plant design, that a new design can be shown to be acceptable for severe accident concerns, in part, by completion of a PRA and consideration of the severe accident vulnerabilities the PRA exposes along with the insights it may add to the assurance of no undue risk to public health and safety.

3. The advanced reactor policy statement indicates the use of PRA as a design tool is implied by the Commission's policy statement on "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (60 FR 42622; August 16, 1995) (Ref. A-13).

Implementation of the LMP methodology inherently conforms to the underlying purposes of these Commission policies.

<u>Supporting Risk-Informed Applications of the PRA in Addition to an Initial Licensing Application</u>

CP applicants may use the PRA to support risk-informed applications, in addition to implementing the LMP methodology, either concurrently with the CP application, concurrently with the OL application, or after the OL issuance. These additional risk-informed applications may affect the PRA scope, level of detail, or elements that should be considered early during PRA development. Examples of additional risk-informed applications include, but are not limited to:

1. Risk-informed inservice inspection and inservice testing programs. Guidance is provided in DANU-ISG-2022-07, "Risk-Informed Inservice Inspection/Inservice Testing Programs for Non-LWRs" (Ref. A-14).

2. Programs to implement the 2019 Edition of the ASME Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," Division 2, "Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants" (Ref. A-15). Guidance is provided in RG 1.246, "Acceptability of ASME Code, Section XI, Division 2, "Requirements for Reliability and Integrity Management (RIM) Programs for Nuclear Power Plants" (Ref. A-16).

3. Risk-informed technical specifications. Guidance is provided in DANU-ISG-2022-08, "Risk-Informed Technical Specifications" (Ref. A-17).

4. Risk-informed fire protection programs. Guidance is provided in DANU-ISG-2022-09, "Risk-Informed, Performance-Based Fire Protection Program (for Operations)" (Ref. A-18).

A.3 PRA Scope

A.3.1 A CP applicant using the LMP methodology should estimate the risk metrics described in Sections C.3.2.1, C.4.1.1, C.4.1.2, and C.4.1.3 of NEI 21-07, Revision 1, on either a qualitative or quantitative basis consistent with the information available when the CP application is prepared. Such an estimation should include an explanation of how the Commission's quantitative health objectives (QHOs) from the Commission Safety Goal Policy Statement will be met in support of the OL application.

A.3.2 Consistent with NEI 21-07, Revision 1, Section 2.1.1, as clarified in Staff Position C.3.c in the main body of this regulatory guidance and RG 1.247 (for trial use), Staff Position C.1.1, the CP applicant should:

1. Identify all radiological sources, hazards, and POSs by performing a comprehensive and systematic search.

2. Disposition the search results by a combination of PRA logic modeling, acceptable screening methods, risk-informed supplemental evaluations, and crediting design-basis hazard levels (DBHLs).

A.3.3 Regarding PRA acceptability, the minimum scope of the CP PRA logic model should include the internal events hazard group for the reactor in the at-power POS. At a minimum, this demonstrates the applicant's ability to develop an acceptable licensing basis and to establish an acceptable foundation for upgrading the PRA logic model as the design progresses.

A.3.4 Regarding PRA acceptability, the high-level requirements[4] and associated supporting requirements defined and stated in ASME/ANS RA-S-1.4-2021, as endorsed in RG 1.247 (for trial use) with clarifications and qualifications, provide an acceptable approach for developing PRA logic models.

A.3.5 Regarding PRA acceptability, Section 4.3.11 of ASME/ANS RA-S-1.4-2021, which is endorsed in RG 1.247 (for trial use) with exceptions, provides an acceptable approach for performing a hazards screening analysis.

A.3.6 Regarding PRA acceptability, risk-informed supplemental evaluations may be used to disposition certain radiological sources, hazards, or POSs. NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking" (Ref. A-19), provides a generally acceptable approach for developing risk-informed supplemental evaluations. Section 1.3 of NUREG-1855 notes that the process described in NUREG-1855 "…is applicable to non-LWRs and reactors in the design stage; however, the screening criteria and the specific sources of uncertainty may not be applicable." Consequently, non-LWR CP applicants who use the guidance in NUREG-1855 to develop risk-informed supplemental evaluations should (1) describe and justify the use of reactor-technology-specific screening criteria, and (2) explain how specific sources of uncertainty were identified and addressed.

A.3.7 The CP applicant may disposition certain hazards by crediting DBHLs in lieu of explicitly modeling these hazards in the PRA or accounting for them through a risk-informed supplementary evaluation. NEI 18-04, Revision 1, Section 3.2.2, Task 6, p. 14, states:

> In many cases, it is expected that the initial selection of SR SSCs [safety-related structures, systems, and components] and selection of the DBAs [design-basis accidents] will be based on a PRA that includes internal events but has not yet been expanded to address external hazards. With the understanding that SR SSCs are required to be capable of performing their RSFs [required safety functions] in response to external events within

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4 The non-LWR PRA standard uses the terms "requirement," "require," and other similar mandatory language. However, the use of this language in this RG does not imply that this RG imposes any regulatory requirement or suggest that these standards are the only way to meet the statutory and regulatory requirements.

the DBEHL [design-basis external hazard levels], there will be no new DBAs introduced by external hazards.

NEI 21-07, Revision 1, Section 6.1.1 clarified NEI 18-04 by defining and using the term DBHL rather than DBEHL. Specifically, DBHLs address both traditional external hazards (e.g., seismic events, external floods, high winds) and internal hazards (e.g., internal fires, internal floods, turbine missiles, and high energy line breaks). Consistent with Section C.1 in RG 1.233, the scope of the PRA, when completed, should cover a full set of internal and external initiating events and provide an estimate of radiological consequences when the design is completed and site characteristics are defined.

A.4 PRA Elements

A.4.1 Regarding PRA acceptability, Table A-1 shows which PRA elements defined in Staff Position C.1.3 of RG 1.247 apply to the minimally acceptable PRA scope and additional PRA elements that may be used to fully implement the LMP methodology at the CP stage.

Table A-1. PRA Elements for Non-LWR CP Applications Based on the LMP Methodology

Minimally Acceptable PRA		Additional PRA Elements	
Identifier	PRA Element	Identifier	PRA Element
C.1.3.2 (IE)	Initiating Event Analysis	C.1.3.1 (POS)	Plant Operating State Analysis
C.1.3.3 (ES)	Event Sequence Analysis	C.1.3.8 (IF)	Internal Flood PRA
C.1.3.4 (SC)	Success Criteria Development	C.1.3.9 (F)	Internal Fire PRA
C.1.3.5 (SY)	Systems Analysis	C.1.3.10 (S)	Seismic PRA
C.1.3.6 (HR)	Human Reliability Analysis	C.1.3.12 (W)	High Wind PRA
C.1.3.7 (DA)	Data Analysis	C.1.3.13 (XF)	External Flooding PRA
C.1.3.11 (HS)	Hazard Screening Analysis	C.1.3.14 (O)	Other Hazards PRA
C.1.3.15 (ESQ)	Event Sequence Quantification		
C.1.3.16 (MS)	Mechanistic Source Term Analysis		
C.1.3.17 (RC)	Radiological Consequence Analysis		
C.1.3.18 (RI)	Risk Integration		

A.4.2 Consistent with Section C.2.1.1 of NEI 21-07, Revision 1, the PRA elements should be developed to conform with the high-level requirements and associated supporting requirements of ASME/ANS RA-S-1.4-2021, as endorsed with clarifications and qualifications in Appendix A of RG 1.247. Consistent with Staff Position C.2.1 in RG 1.247, all high-level requirements for a given PRA element should be met.

A.5 PRA Level of Detail

A.5.1 Consistent with Section C.2.1.1 of NEI 21-07, Revision 1, the CP PRA level of detail should be commensurate with the preliminary plant design and site characteristics described in the PSAR.

A.5.2 The level of detail in a CP PRA should be established using the process provided in Section 3 of ASME/ANS RA-S-1.4-2021, “Risk Assessment Application Process.” If an applicant meets the provisions of Table A-2, the NRC staff would consider this to result in an acceptable level of detail for the PRA logic model and hazard screening analyses supporting an LMP-based CP application. To the extent the provisions in Table A-2 cannot be met due to the maturity of preliminary plant design and information about site characteristics, the application should justify the adequacy of the internal events PRA logic model for the reactor in the at-power POSs to support the implementation of the LMP methodology. If the maturity of the preliminary plant design and information about site characteristics is sufficient to support the development of a PRA logic model that includes other internal and external hazards or other POSs, the applicant should consider applying the provisions of Table A-3.

A.6 Plant Representation and PRA Configuration Control

A.6.1 Consistent with Section C.2.1.1 of NEI 21-07, Revision 1, the CP applicant should establish a PRA configuration control program to ensure that the CP PRA reasonably represents the preliminary plant design and site characteristics described in the PSAR.

A.6.2 Regarding PRA acceptability, Section 5 of ASME/ANS RA-S-1.4-2021, which is endorsed in RG 1.247 (for trial use) with exceptions, provides one acceptable approach for establishing a PRA configuration control program.

A.6.3 Regarding PRA acceptability, consistent with the discussion provided in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (Ref. A-20), Section 19.0, Item 9, page 19.0-12, the PRA configuration control program may be a stand-alone program or included within the quality assurance program required by 10 CFR 50.34(a)(7).

A.7 PRA Documentation

A.7.1 NEI 21-07, Revision 1, as endorsed with additions and clarifications in the main body of this RG, provides an acceptable approach and format for providing CP PRA information in the SAR. Consistent with staff position C.4.2 of RG 1.247 on PRA documentation for an application, the PSAR should provide a summary justification of the acceptability of the CP PRA. This justification should summarize why, commensurate with the level of maturity of the facility design, the scope, level of detail, elements of the PRA, and plant representation described in the CP PRA is sufficient to implement the LMP methodology for the CP application. To this end, the summary justification should describe, but is not limited to, the following:

- How the CP PRA was developed in accordance with the guidance provided in this appendix and the portions of RG 1.247 and the ASME/ANS non-LWR PRA consensus standard (ASME/ANS RA-S-1.4-2021) the applicant has followed in developing the CP PRA.

- How a PRA configuration control program has been used to ensure the CP PRA represents the as-designed, as-to-be-built, as-to-be-operated facility described in the CP application.

- How the PRA configuration control program will ensure that the PRA (i.e., PRA logic model, screening analyses, and risk-informed supplemental evaluations) supporting the OL application will represent the as-built, as-to-be-operated facility; account for all radiological sources, all hazards, and all plant operating states; and meet all applicable staff positions in RG 1.247 and technical elements in ASME/ANS RA-S-1.4-2021 that the CP PRA did not meet.

- How the applicant's self-assessment of the CP PRA was performed consistent with the related staff positions in Section A.8 of this regulatory guide.

A.7.2 Regarding acceptability of archival documentation confirming the acceptability of the CP PRA, Staff Position C.4.1 in RG 1.247 provides an acceptable approach for developing and preserving PRA archival information. PRA documentation providing the detailed justification for the acceptability of the PRA should be maintained in archival PRA documentation and, at a minimum, should include the items described in staff position C.4.1 of RG 1.247.

A.7.3 Consistent with the discussion provided in NUREG-0800, Section 19.0, Item 9, page 19.0-12, PRA archival information may be controlled by a stand-alone program, or the quality assurance program required by 10 CFR 50.34(a)(7).

A.8 Demonstrating PRA Acceptability

A.8.1 The guidance in DANU-ISG-2022-05, "Organization and Human-System Considerations" (Ref. A-21), Section 11.1.1, provides an acceptable approach for describing key management responsibilities for developing the PRA.

A.8.2 The guidance in DANU-ISG-2022-05, Section 11.1.1.1, provides an acceptable approach for describing the ability of the CP applicant's technical staff to develop the PRA.

A.8.3 Regarding PRA acceptability, the CP applicant should conduct a self-assessment to demonstrate that all PRA logic models, screening analyses, and risk-informed supplemental analyses have been developed and used in a technically acceptable manner, including the appropriateness of assumptions and approximations. The self-assessment, thus, should provide a basis for asserting that the CP PRA is acceptable for implementing the LMP methodology leading up to submittal of the CP application. To this end, the self-assessment should review:

1. The comprehensive and systematic search used to identify radiological sources, POSs, and hazards.

2. The PRA logic models (including the scope, level of detail, and elements), screening analyses, risk-informed supplemental evaluations, and credit for DBHLs.

3. The CP applicant's PRA configuration control program used to ensure that CP PRA logic models, screening analyses, and risk-informed supplemental analyses represent the as-designed, as-to-be-built, and as-to-be-operated plant.

A.8.4 Regarding PRA acceptability, the guidance in NEI 20-09, Revision 1, "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard," (Ref. A-22), Sections 3.2, A.3.1, and A.3.2, which is endorsed in RG 1.247 with no exceptions, provides an acceptable approach for performing a self-assessment.

A.8.5 Regarding PRA acceptability, in addition to a self-assessment, a CP applicant may have the comprehensive and systematic search, some or all PRA elements, some or all screening analyses, or the PRA configuration control program peer reviewed prior to submittal of the CP application.[5] Section 6 of ASME/ANS RA-S-1.4-2021, which is endorsed in Staff Position C.2.2 in RG 1.247 with exceptions, and NEI 20-09, Revision 1, which is endorsed in RG 1.247 without exception, provide an acceptable approach for performing a peer review.

The purpose of a peer review is to determine whether the relevant high-level requirements and associated supporting requirements established in ASME/ANS RA-S-1.4-2021, as endorsed in RG 1.247 with exceptions, have been met. The peer review should also confirm that the technical aspects of the CP PRA have been developed in a technically correct manner and assess the appropriateness of assumptions and

5 High-level requirements and supporting requirements related to the comprehensive and systematic search for radionuclide sources, POSs, and hazards appear in various locations throughout ASME/ANS RA-S-1.4-2021. However, the standard does not provide high-level requirements and supporting requirements for risk-informed supplemental evaluations.

approximations used in the CP PRA. As a result, completion of a peer review may reduce the need for an in-depth staff review of the CP PRA.

Appendix A Tables A-2 and A-3 Notes

The staff applied the risk assessment application process provided in Section 3 of ASME/ANS RA-S-1.4-2021 to determine the applicability of supporting requirements, as defined in the ASME/ANS RA-S-1.4-2021, for a non-LWR CP application that implements the LMP methodology. The staff's application of this process considered a wide range of maturities of design information that may be submitted to the NRC in a CP application. The results, which reflect the provisions of ASME/ANS RA-S-1.4-2021 that an applicant should follow for a fully mature design, are presented in Tables A-2 and A-3, which follow immediately below. The CP applicant should use Tables A-2 and A-3 as guidance to demonstrate the acceptability of the scope and level of detail of the PRA logic model, consistent with the maturity of the design. Alternatively, the CP applicant may perform a separate analysis using the process provided in Section 3 of ASME/ANS RA-S-1.4-2021 and justify any departures from or alternatives to Tables A-2 and A-3, depending on the maturity of design information for a given applicant. As such, an applicant need not use Tables A-2 and A-3 if an applicant justifies a different combination of applicable ASME/ANS supporting requirements for its application.

As referenced in staff position A.5.2, Table A-2 shows the alphanumeric identifiers of the underlined high-level requirements and their related supporting requirements in tabular format from ASME/ANS RA-S-1.4-2021 for a minimally acceptable PRA logic model and hazard screening analysis for a CP application under 10 CFR Part 50. As discussed in position A.3.3, the minimally acceptable scope of the CP PRA logic model should include the internal events hazard for the reactor in the at-power POS. Table A-3 shows the related alphanumeric identifiers for a PRA logic model associated with a design that includes other internal and external hazards or other POSs for a CP application. The in-line entry associated with each ASME/ANS supporting requirement is either "Yes," "CC-I," or "CC-II." A "Yes" entry indicates the supporting requirements for CC-I and CC-II (i.e., the two types of capability categories in ASME/ANS RA-S-1.4-2021) are applicable because the supporting requirement is the same for both capability categories. An entry of either "CC-I" or "CC-II" indicates the minimum applicable supporting requirement associated with that capability category. The staff did not include inapplicable supporting requirements in these tables.

Table A-2. Non-LWR CP Applications Based on the LMP Methodology: Applicability of ASME/ANS Non-LWR PRA Standard High-Level Requirements and Supporting Requirements to PRA Elements Used to Develop a Minimally Acceptable PRA (1 of 4)

C.1.3.2 (IE)		HLR-IE-D		ES-C9	CC-II	SY-A4	Yes
HLR-IE-A		IE-D1	Yes	ES-C10	Yes	SY-A6	CC-II
IE-A1	Yes	IE-D2	Yes	ES-C11	Yes	SY-A7	CC-I
IE-A2	Yes	IE-D3	Yes			SY-A8	Yes
IE-A4	Yes			HLR-ES-D		SY-A9	Yes
IE-A5	Yes	**C.1.3.3 (ES)**		ES-D1	Yes	SY-A11	Yes
IE-A6	Yes	HLR-ES-A		ES-D2	Yes	SY-A12	Yes
IE-A8	CC-II	ES-A1	Yes	ES-D3	Yes	SY-A13	Yes
IE-A9	CC-II	ES-A2	Yes			SY-A14	Yes
IE-A10	CC-I	ES-A3	Yes	**C.1.3.4 (SC)**		SY-A15	Yes
IE-A11	Yes	ES-A4	Yes	HLR-SC-A		SY-A16	Yes
IE-A12	CC-I	ES-A5	Yes	SC-A1	Yes	SY-A17	Yes
IE-A14	Yes	ES-A6	Yes	SC-A2	Yes	SY-A18	Yes
IE-A15	Yes	ES-A7	Yes	SC-A3	CC-II	SY-A19	Yes
IE-A16	Yes	ES-A8	Yes	SC-A4	Yes	SY-A20	Yes
IE-A17	Yes	ES-A9	Yes	SC-A5	Yes	SY-A21	Yes
IE-A18	Yes	ES-A10	CC-II	SC-A6	Yes	SY-A22	Yes
		ES-A11	Yes	SC-A7	CC-II	SY-A23	Yes
HLR-IE-B		ES-A12	CC-II	SC-A8	Yes	SY-A24	Yes
IE-B1	Yes	ES-A13	Yes	SC-A9	Yes	SY-A25	Yes
IE-B2	Yes	ES-A14	Yes	SC-A10	Yes	SY-A26	Yes
IE-B3	Yes	ES-A15	Yes	SC-A11	Yes	SY-A27	Yes
IE-B4	CC-II					SY-A28	Yes
IE-B5	Yes	HLR-ES-B		HLR-SC-B		SY-A29	CC-I
IE-B6	Yes	ES-B1	Yes	SC-B1	CC-II	SY-A30	Yes
		ES-B2	Yes	SC-B2	Yes	SY-A31	Yes
HLR-IE-C		ES-B3	Yes	SC-B3	Yes	SY-A32	Yes
IE-C2	Yes	ES-B4	Yes	SC-B4	Yes	SY-A33	Yes
IE-C4	Yes	ES-B5	Yes	SC-B5	Yes		
IE-C5	Yes	ES-B6	Yes	SC-B6	Yes	HLR-SY-B	
IE-C7	Yes	ES-B7	Yes	SC-B7	CC-I	SY-B1	CC-I
IE-C8	Yes	ES-B8	Yes	SC-B8	Yes	SY-B2	CC-I
IE-C9	Yes	ES-B9	Yes	SC-B9	Yes	SY-B3	Yes
IE-C10	CC-II	ES-B10	Yes	SC-B10	Yes	SY-B4	Yes
IE-C11	Yes					SY-B5	Yes
IE-C12	Yes	HLR-ES-C		HLR-SC-C		SY-B6	Yes
IE-C13	Yes	ES-C1	Yes	SC-C1	Yes	SY-B7	CC-I
IE-C14	Yes	ES-C2	Yes	SC-C2	Yes	SY-B8	Yes
IE-C15	Yes	ES-C3	Yes	SC-C3	Yes	SY-B9	Yes
IE-C16	Yes	ES-C4	Yes			SY-B10	Yes
IE-C17	CC-II	ES-C5	Yes	**C.1.3.5 (SY)**		SY-B11	CC-II
IE-C18	Yes	ES-C6	Yes	HLR-SY-A		SY-B12	Yes
IE-C19	CC-II	ES-C7	CC-I	SY-A1	Yes	SY-B13	Yes
		ES-C8	Yes	SY-A2	Yes	SY-B14	Yes
				SY-A3	Yes	SY-B15	Yes

Table A-2. Non-LWR CP Applications Based on the LMP Methodology: Applicability of ASME/ANS Non-LWR PRA Standard High-Level Requirements and Supporting Requirements to PRA Elements Used to Develop a Minimally Acceptable PRA (2 of 4)

SY-B16	Yes
SY-B17	Yes
HLR-SY-C	
SY-C1	Yes
SY-C2	Yes
SY-C3	Yes
C.1.3.6 (HR)	
HLR-HR-A	
HR-A1	Yes
HR-A2	Yes
HR-A3	Yes
HR-A4	Yes
HR-A5	Yes
HR-A6	Yes
HR-A7	Yes
HR-A8	Yes
HR-A9	Yes
HR-A10	Yes
HLR-HR-B	
HR-B1	Yes
HR-B2	Yes
HR-B3	Yes
HLR-HR-C	
HR-C1	Yes
HR-C2	Yes
HR-C3	Yes
HR-C4	CC-II
HR-C5	Yes
HR-C6	Yes
HLR-HR-D	
HR-D1	Yes
HR-D2	CC-I
HR-D3	Yes
HR-D4	CC-I
HR-D5	Yes
HR-D7	Yes
HR-D8	CC-II
HR-D9	Yes
HR-D10	Yes
HLR-HR-E	
HR-E2	Yes
HR-E3	Yes
HR-E4	Yes
HR-E6	CC-I
HR-E7	CC-I
HR-E8	Yes
HR-E9	Yes
HLR-HR-F	
HR-F1	Yes
HR-F2	Yes
HR-F3	CC-II
HR-F4	CC-II
HR-F5	Yes
HLR-HR-G	
HR-G1	CC-I
HR-G2	Yes
HR-G3	Yes
HR-G4	CC-I
HR-G5	CC-II
HR-G6	CC-II
HR-G7	Yes
HR-G8	CC-I
HR-G10	Yes
HR-G11	Yes
HR-G12	Yes
HR-G13	Yes
HR-G14	CC-II
HR-G15	Yes
HR-G16	Yes
HLR-HR-H	
HR-H1	Yes
HR-H2	Yes
HR-H3	Yes
HR-H4	CC-I
HR-H5	Yes
HR-H6	Yes
HLR-HR-I	
HR-I1	Yes
HR-I2	Yes
HR-I3	Yes
C.1.3.7 (DA)	
HLR-DA-A	
DA-A1	Yes
DA-A2	Yes
DA-A3	Yes
DA-A4	Yes
DA-A5	Yes
DA-A6	Yes
HLR-DA-B	
DA-B1	CC-II
DA-B2	Yes
HLR-DA-C	
DA-C1	Yes
DA-C2	Yes
DA-C9	CC-I
DA-C14	Yes
DA-C15	Yes
DA-C17	CC-I
DA-C19	Yes
DA-C20	Yes
DA-C21	Yes
DA-C23	Yes
DA-C25	Yes
HLR-DA-D	
DA-D1	CC-I
DA-D2	Yes
DA-D3	CC-II
DA-D5	Yes
DA-D6	Yes
DA-D7	CC-II
DA-D8	CC-I
DA-D9	Yes
HLR-DA-E	
DA-E1	Yes
DA-E2	Yes
DA-E3	Yes
C.1.3.11 (HS)	
HLR-HS-A	
HS-A1	Yes
HS-A2	Yes
HS-A3	Yes
HS-A4	Yes
HLR-HS-B	
HS-B1	Yes
HS-B2	Yes
HS-B4	Yes
HS-B5	Yes
HS-B6	Yes
HS-B7	Yes
HLR-HS-C	
HS-C1	Yes
HS-C2	Yes
HS-C3	Yes
HS-C4	Yes
HS-C5	Yes
HS-C6	Yes
HS-C7	Yes
HS-C8	Yes
HS-C9	Yes
HS-C10	Yes
HS-C11	Yes
HS-C12	Yes
HS-C13	Yes
HS-C14	Yes
HLR-HS-D	
HS-D1	Yes
HLR-HS-E	
HS-E1	Yes
HS-E2	Yes
HS-E3	Yes
C.1.3.15 (ESQ)	
HLR-ESQ-A	
ESQ-A1	Yes
ESQ-A2	Yes
ESQ-A3	Yes
ESQ-A4	Yes
ESQ-A5	CC-II
ESQ-A6	Yes
ESQ-A7	Yes

Table A-2. Non-LWR CP Applications Based on the LMP Methodology: Applicability of ASME/ANS Non-LWR PRA Standard High-Level Requirements and Supporting Requirements to PRA Elements Used to Develop a Minimally Acceptable PRA (3 of 4)

ESQ-A8	CC-I	HLR-ESQ-E		HLR-MS-E		HLR-RCPA-C	
ESQ-A9	CC-I	ESQ-E1	Yes	MS-E1	Yes	RCPA-C1	Yes
		ESQ-E2	CC-II	MS-E2	Yes	RCPA-C2	Yes
HLR-ESQ-B				MS-E3	Yes		
ESQ-B1	Yes	HLR-ESQ-F		MS-E4	Yes	HLR-RCME-A	
ESQ-B2	Yes	ESQ-F1	Yes			RCME-A1	Yes
ESQ-B3	Yes	ESQ-F2	Yes	**C.1.3.17 (RC)**		RCME-A2	CC-I
ESQ-B4	Yes	ESQ-F3	Yes	HLR-RCRE-A		RCME-A3	CC-II
ESQ-B5	Yes	ESQ-F4	Yes	RCRE-A1	Yes	RCME-A4	CC-II
ESQ-B6	Yes	ESQ-F5	Yes	RCRE-A2	Yes	RCME-A5	CC-II
ESQ-B7	Yes			RCRE-A3	Yes	RCME-A6	CC-II
ESQ-B8	Yes	**C.1.3.16 (MS)**				RCME-A7	CC-II
ESQ-B9	Yes	HLR-MS-A		HLR-RCRE-B		RCME-A8	Yes
ESQ-B10	Yes	MS-A1	Yes	RCRE-B1	Yes	RCME-A9	Yes
		MS-A2	Yes	RCRE-B2	Yes	RCME-A10	Yes
HLR-ESQ-C		MS-A3	CC-II				
ESQ-C1	Yes	MS-A4	Yes	HLR-RCRE-C		HLR-RCME-B	
ESQ-C2	Yes	MS-A5	Yes	RCRE-C1	Yes	RCME-B1	Yes
ESQ-C3	Yes					RCME-B2	Yes
ESQ-C4	Yes	HLR-MS-B		HLR-RCPA-A			
ESQ-C5	CC-II	MS-B1	CC-I	RCPA-A1	Yes	HLR-RCAD-A	
ESQ-C6	CC-II	MS-B2	Yes	RCPA-A2	Yes	RCAD-A1	CC-II
ESQ-C7	CC-I	MS-B3	Yes	RCPA-A3	Yes	RCAD-A2	CC-II
ESQ-C8	CC-I	MS-B4	Yes	RCPA-A4	CC-II	RCAD-A3	CC-II
ESQ-C9	CC-I	MS-B5	Yes	RCPA-A5	Yes	RCAD-A4	Yes
ESQ-C10	Yes	MS-B6	Yes	RCPA-A6	Yes	RCAD-A5	Yes
ESQ-C11	Yes	MS-B7	Yes	RCPA-A7	Yes	RCAD-A6	Yes
ESQ-C12	Yes			RCPA-A8	Yes	RCAD-A7	Yes
ESQ-C13	Yes	HLR-MS-C		RCPA-A9	Yes		
ESQ-C14	CC-I	MS-C1	CC-I	RCPA-A10	Yes	HLR-RCAD-B	
ESQ-C15	CC-II	MS-C2	CC-I	RCPA-A11	Yes	RCAD-B1	Yes
ESQ-C16	Yes	MS-C3	CC-I	RCPA-A12	Yes	RCAD-B2	CC-I
ESQ-C17	Yes	MS-C4	CC-II	RCPA-A13	Yes		
		MS-C5	Yes			HLR-RCAD-C	
HLR-ESQ-D		MS-C6	Yes	HLR-RCPA-B		RCAD-C1	Yes
ESQ-D1	Yes	MS-C7	Yes	RCPA-B1	CC-II	RCAD-C2	CC-II
ESQ-D2	Yes			RCPA-B2	CC-II	RCAD-C3	Yes
ESQ-D3	Yes	HLR-MS-D		RCPA-B3	CC-II	RCAD-C4	Yes
ESQ-D4	CC-II	MS-D1	Yes	RCPA-B4	Yes	RCAD-C5	Yes
ESQ-D5	Yes	MS-D2	CC-II	RCPA-B6	Yes		
ESQ-D6	CC-II	MS-D3	Yes	RCPA-B7	Yes	HLR-RCAD-D	
ESQ-D7	Yes	MS-D4	CC-II			RCAD-D1	CC-II
ESQ-D8	Yes					RCAD-D2	CC-II
						RCAD-D3	Yes

Table A-2. Non-LWR CP Applications Based on the LMP Methodology: Applicability of ASME/ANS Non-LWR PRA Standard High-Level Requirements and Supporting Requirements to PRA Elements Used to Develop a Minimally Acceptable PRA (4 of 4)			
<u>HLR-RCAD-E</u> RCAD-E1 CC-II RCAD-E2 CC-II RCAD-E3 CC-II RCAD-E4 CC-II RCAD-E5 Yes RCAD-E6 Yes <u>HLR-RCAD-F</u> RCAD-F1 Yes RCAD-F2 Yes <u>HLR-RCDO-A</u> RCDO-A1 Yes RCDO-A2 Yes RCDO-A3 Yes RCDO-A4 CC-II RCDO-A5 Yes RCDO-A6 Yes RCDO-A7 CC-II RCDO-A8 Yes RCDO-A9 Yes RCDO-A10 Yes <u>HLR-RCDO-B</u> RCDO-B1 CC-II RCDO-B2 Yes <u>HLR-RCDO-C</u> RCDO-C1 Yes RCDO-C2 Yes <u>HLR-RCHE-A</u> RCHE-A1 Yes RCHE-A2 CC-I RCHE-A3 CC-I RCHE-A4 Yes RCHE-A5 Yes RCHE-A6 Yes <u>HLR-RCHE-B</u> RCHE-B1 Yes RCHE-B2 Yes RCHE-B3 Yes	<u>HLR-RCHE-C</u> RCHE-C1 Yes RCHE-C2 Yes <u>HLR-RCQ-A</u> RCQ-A1 Yes RCQ-A2 Yes RCQ-A3 Yes <u>HLR-RCQ-B</u> RCQ-B1 Yes RCQ-B2 Yes RCQ-B3 Yes <u>HLR-RCQ-C</u> RCQ-C1 Yes RCQ-C2 CC-I <u>HLR-RCQ-D</u> RCQ-D1 Yes RCQ-D2 Yes RCQ-D3 Yes **C.1.3.18 (RI)** <u>HLR-RI-A</u> RI-A1 Yes RI-A2 Yes RI-A3 Yes RI-A4 Yes RI-A5 Yes <u>HLR-RI-B</u> RI-B1 Yes RI-B2 CC-II RI-B3 CC-II RI-B4 Yes RI-B5 Yes RI-B6 Yes RI-B7 Yes <u>HLR-RI-C</u> RI-C1 Yes RI-C2 Yes RI-C3 Yes RI-C4 CC-II	<u>HLR-RI-D</u> RI-D1 Yes RI-D2 Yes	

Table A-3. Non-LWR CP Applications Based on the LMP Methodology: Applicability of ASME/ANS Non-LWR PRA Standard High-Level Requirements and Supporting Requirements to Additional PRA Elements (1 of 6)

C.1.3.1 (POS)		FLPP-B7	Yes	FLSN-A20	Yes	HLR-FLPR-C	
HLR-POS-A		FLPP-B8	Yes	FLSN-A21	Yes	FLPR-C1	Yes
POS-A1	CC-II					FLPR-C2	Yes
POS-A2	Yes	HLR-FLPP-C		HLR-FLSN-B		FLPR-C3	Yes
POS-A3	Yes	FLPP-C1	Yes	FLSN-B1	Yes		
POS-A5	Yes	FLPP-C2	Yes	FLSN-B2	Yes	HLR-FLHR-A	
POS-A8	Yes	FLPP-C3	Yes	FLSN-B3	Yes	FLHR-A1	CC-I
POS-A9	Yes					FLHR-A2	CC-I
POS-A10	Yes	HLR-FLSO-A		HLR-FLEV-A			
POS-A11	Yes	FLSO-A1	Yes	FLEV-A1	CC-II	HLR-FLHR-B	
POS-A12	Yes	FLSO-A2	Yes	FLEV-A2	CC-I	FLHR-B1	Yes
POS-A13	Yes	FLSO-A3	Yes	FLEV-A3	Yes	FLHR-B2	CC-II
		FLSO-A4	Yes	FLEV-A4	Yes	FLHR-B3	CC-II
HLR-POS-B		FLSO-A5	Yes				
POS-B1	CC-II	FLSO-A6	Yes	HLR-FLEV-B		HLR-FLHR-C	
POS-B2	Yes	FLSO-A7	Yes	FLEV-B1	CC-I	FLHR-C1	CC-II
POS-B3	Yes	FLSO-A8	Yes	FLEV-B2	Yes		
POS-B4	Yes	FLSO-A9	Yes	FLEV-B3	CC-I	HLR-FLHR-D	
POS-B5	CC-I			FLEV-B4	CC-I	FLHR-D1	CC-I
POS-B6	Yes	HLR-FLSO-B		FLEV-B5	Yes	FLHR-D2	Yes
POS-B7	Yes	FLSO-B1	Yes	FLEV-B6	Yes	FLHR-D3	Yes
POS-B8	Yes	FLSO-B2	Yes	FLEV-B7	Yes		
		FLSO-B3	Yes			HLR-FLHR-E	
HLR-POS-C				HLR-FLEV-C		FLHR-E1	Yes
POS-C1	Yes	HLR-FLSN-A		FLEV-C1	Yes	FLHR-E2	Yes
POS-C2	Yes	FLSN-A1	Yes	FLEV-C2	Yes	FLHR-E3	Yes
POS-C3	Yes	FLSN-A2	Yes	FLEV-C3	Yes		
POS-C4	Yes	FLSN-A3	Yes			HLR-FLESQ-A	
		FLSN-A4	Yes	HLR-FLPR-A		FLESQ-A1	Yes
HLR-POS-D		FLSN-A5	Yes	FLPR-A1	Yes	FLESQ-A2	Yes
POS-D1	Yes	FLSN-A6	CC-II	FLPR-A2	Yes	FLESQ-A3	CC-I
POS-D2	Yes	FLSN-A7	Yes	FLPR-A3	Yes	FLESQ-A4	CC-I
POS-D3	Yes	FLSN-A8	CC-II			FLESQ-A5	Yes
		FLSN-A9	CC-I	HLR-FLPR-B		FLESQ-A6	Yes
C.1.3.8 (FL)		FLSN-A10	CC-I	FLPR-B1	Yes	FLESQ-A7	Yes
HLR-FLPP-A		FLSN-A11	Yes	FLPR-B2	Yes	FLESQ-A8	Yes
FLPP-A1	Yes	FLSN-A12	Yes	FLPR-B3	CC-II		
		FLSN-A13	Yes	FLPR-B4	CC-II	HLR-FLESQ-B	
HLR-FLPP-B		FLSN-A14	Yes	FLPR-B5	CC-II	FLESQ-B1	Yes
FLPP-B1	Yes	FLSN-A15	Yes	FLPR-B6	CC-I		
FLPP-B2	Yes	FLSN-A16	Yes	FLPR-B7	CC-I	HLR-FLESQ-C	
FLPP-B4	Yes	FLSN-A17	Yes	FLPR-B8	CC-II	FLESQ-C1	Yes
FLPP-B5	Yes	FLSN-A18	Yes	FLPR-B9	Yes		
FLPP-B6	Yes	FLSN-A19	Yes	FLPR-B10	Yes		

Table A-3. Non-LWR CP Applications Based on the LMP Methodology: Applicability of ASME/ANS Non-LWR PRA Standard High-Level Requirements and Supporting Requirements to Additional PRA Elements (2 of 6)

HLR-FLESQ-D	
FLESQ-D1	CC-II
HLR-FLESQ-E	
FLESQ-E1	Yes
FLESQ-E2	CC-II
HLR-FLESQ-F	
FLESQ-F1	CC-II
FLESQ-F2	Yes
FLESQ-F3	Yes
FLESQ-F4	Yes
FLESQ-F5	Yes
C.1.3.9 (F)	
HLR-FPP-A	
FPP-A1	Yes
HLR-FPP-B	
FPP-B1	Yes
FPP-B2	Yes
FPP-B3	Yes
FPP-B4	Yes
FPP-B6	Yes
FPP-B7	Yes
FPP-B8	Yes
HLR-FPP-C	
FPP-C1	Yes
FPP-C2	Yes
FPP-C3	Yes
HLR-FES-A	
FES-A1	Yes
FES-A2	Yes
FES-A3	Yes
FES-A4	Yes
FES-A5	CC-II
FES-A6	CC-II
FES-A7	Yes
HLR-FES-B	
FES-B1	Yes
FES-B2	CC-II
FES-B3	Yes
HLR-FES-C	
FES-C1	CC-I
FES-C2	Yes
FES-C3	Yes
HLR-FES-D	
FES-D1	Yes
FES-D2	Yes
FES-D3	Yes
HLR-FCS-A	
FCS-A1	CC-I
FCS-A2	Yes
FCS-A3	CC-I
FCS-A4	Yes
HLR-FCS-B	
FCS-B1	Yes
FCS-B2	Yes
FCS-B3	Yes
HLR-FCS-C	
FCS-C1	Yes
FCS-C2	Yes
FCS-C3	Yes
HLR-FQLS-A	
FQLS-A1	Yes
FQLS-A2	Yes
FQLS-A3	Yes
FQLS-A4	Yes
FQLS-A5	Yes
FQLS-A6	Yes
HLR-FQLS-B	
FQLS-B1	Yes
FQLS-B2	Yes
FQLS-B3	Yes
HLR-FPRM-A	
FPRM-A1	Yes
FPRM-A2	Yes
FPRM-A3	Yes
HLR-FPRM-B	
FPRM-B1	Yes
FPRM-B2	Yes
FPRM-B4	Yes
FPRM-B5	CC-II
FPRM-B6	CC-II
FPRM-B7	CC-II
FPRM-B8	CC-II
FPRM-B9	CC-I
FPRM-B10	CC-I
FPRM-B11	Yes
FPRM-B12	Yes
FPRM-B13	CC-I
FPRM-B14	Yes
FPRM-B15	Yes
FPRM-B16	Yes
FPRM-B17	Yes
HLR-FPRM-C	
FPRM-C1	Yes
FPRM-C2	Yes
FPRM-C3	Yes
FPRM-C4	Yes
HLR-FSS-A	
FSS-A1	Yes
FSS-A2	Yes
FSS-A3	Yes
FSS-A4	Yes
HLR-FSS-B	
FSS-B1	Yes
FSS-B2	CC-I
HLR-FSS-C	
FSS-C1	CC-I
FSS-C2	CC-I
FSS-C3	Yes
FSS-C4	CC-I
FSS-C5	Yes
FSS-C6	CC-I
FSS-C7	Yes
HLR-FSS-D	
FSS-D1	Yes
FSS-D2	CC-I
FSS-D3	Yes
FSS-D4	Yes
FSS-D5	Yes
FSS-D6	Yes
FSS-D7	Yes
FSS-D8	Yes
FSS-D9	Yes
FSS-D11	Yes
HLR-FSS-E	
FSS-E1	CC-I
FSS-E2	Yes
FSS-E3	Yes
FSS-E4	CC-II
FSS-E5	Yes
HLR-FSS-F	
FSS-F1	Yes
FSS-F2	CC-I
HLR-FSS-G	
FSS-G1	Yes
FSS-G2	Yes
FSS-G3	Yes
FSS-G4	CC-II
FSS-G5	CC-II
FSS-G6	CC-II
FSS-G7	CC-II
FSS-G8	Yes
FSS-G9	Yes
HLR-FSS-H	
FSS-H1	Yes
FSS-H2	Yes
FSS-H3	Yes
FSS-H4	Yes

Table A-3. Non-LWR CP Applications Based on the LMP Methodology: Applicability of ASME/ANS Non-LWR PRA Standard High-Level Requirements and Supporting Requirements to Additional PRA Elements (3 of 6)

HLR-FIGN-A	
FIGN-A1	Yes
FIGN-A2	Yes
FIGN-A3	Yes
FIGN-A6	Yes
FIGN-A7	CC-I
FIGN-A8	Yes
FIGN-A9	Yes
FIGN-A10	CC-II
FIGN-A11	Yes
FIGN-A12	Yes
HLR-FIGN-B	
FIGN-B1	Yes
FIGN-B2	Yes
FIGN-B3	Yes
HLR-FCF-A	
FCF-A1	CC-I
FCF-A2	CC-II
FCF-A3	Yes
FCF-A4	Yes
HLR-FCF-B	
FCF-B1	Yes
FCF-B2	Yes
FCF-B3	Yes
HLR-FHR-A	
FHR-A1	CC-I
FHR-A3	CC-II
HLR-FHR-B	
FHR-B1	CC-II
FHR-B2	CC-II
HLR-FHR-C	
FHR-C1	CC-I
HLR-FHR-D	
FHR-D1	Yes
FHR-D2	Yes
FHR-D3	Yes
HLR-FHR-E	
FHR-E1	Yes
FHR-E2	Yes
FHR-E3	Yes
HLR-FESQ-A	
FESQ-A1	Yes
FESQ-A2	Yes
FESQ-A3	Yes
FESQ-A4	Yes
FESQ-A5	CC-I
HLR-FESQ-B	
FESQ-B1	Yes
HLR-FESQ-C	
FESQ-C1	CC-I
HLR-FESQ-D	
FESQ-D1	CC-II
FESQ-D2	Yes
FESQ-D3	Yes
HLR-FESQ-E	
FESQ-E1	Yes
FESQ-E2	CC-II
HLR-FESQ-F	
FESQ-F1	Yes
FESQ-F2	Yes
FESQ-F3	Yes
FESQ-F4	Yes
C.1.3.10 (S)	
HLR-SHA-A	
SHA-A1	Yes
SHA-A2	Yes
SHA-A3	Yes
SHA-A4	Yes
SHA-A5	Yes
SHA-A6	Yes
SHA-A7	Yes
HLR-SHA-B	
SHA-B1	Yes
SHA-B2	Yes
SHA-B3	Yes
SHA-B4	Yes
SHA-B5	Yes
HLR-SHA-C	
SHA-C1	Yes
SHA-C2	Yes
SHA-C3	Yes
SHA-C4	Yes
SHA-C5	Yes
HLR-SHA-D	
SHA-D1	Yes
SHA-D2	Yes
SHA-D3	Yes
SHA-D4	Yes
HLR-SHA-E	
SHA-E1	Yes
SHA-E3	Yes
SHA-E5	Yes
HLR-SHA-F	
SHA-F1	Yes
SHA-F2	Yes
SHA-F3	Yes
SHA-F4	Yes
HLR-SHA-G	
SHA-G1	Yes
SHA-G2	Yes
HLR-SHA-H	
SHA-H1	Yes
SHA-H2	Yes
SHA-H3	Yes
SHA-H4	Yes
HLR-SHA-I	
SHA-I1	Yes
SHA-I2	Yes
SHA-I3	Yes
HLR-SFR-A	
SFR-A1	Yes
SFR-A2	Yes
HLR-SFR-B	
SFR-B1	CC-I
SFR-B2	Yes
SFR-B3	Yes
SFR-B4	CC-II
SFR-B5	CC-II
SFR-B6	Yes
HLR-SFR-C	
SFR-C1	Yes
SFR-C2	Yes
HLR-SFR-D	
SFR-D1	Yes
SFR-D2	Yes
SFR-D4	CC-I
SFR-D5	Yes
SFR-D6	Yes
SFR-D7	Yes
SFR-D8	Yes
HLR-SFR-E	
SFR-E1	CC-I
SFR-E2	CC-I
SFR-E3	CC-I
SFR-E4	CC-I
SFR-E5	CC-I
SFR-E6	Yes
SFR-E7	Yes
HLR-SFR-F	
SFR-F1	Yes
SFR-F2	Yes
SFR-F3	Yes
HLR-SPR-A	
SPR-A1	Yes
SPR-A2	Yes

Table A-3. Non-LWR CP Applications Based on the LMP Methodology: Applicability of ASME/ANS Non-LWR PRA Standard High-Level Requirements and Supporting Requirements to Additional PRA Elements (4 of 6)

SPR-A3	Yes	SPR-F2	Yes	WHA-F2	Yes	WFR-E3	Yes
SPR-A4	Yes	SPR-F3	Yes	WHA-F3	CC-II	WFR-E4	Yes
		SPR-F4	Yes	WHA-F4	CC-II	WFR-E5	Yes
HLR-SPR-B		SPR-F5	Yes			WFR-E6	Yes
SPR-B1	Yes			HLR-WHA-G		WFR-E7	Yes
SPR-B2	Yes	**C.1.3.12 (W)**		WHA-G1	Yes	WFR-E8	CC-II
SPR-B3	Yes	HLR-WHA-A		WHA-G2	Yes	WFR-E9	CC-II
SPR-B4	Yes	WHA-A1	Yes	WHA-G3	Yes	WFR-E10	Yes
SPR-B5	Yes	WHA-A2	Yes			WFR-E11	Yes
SPR-B6	Yes	WHA-A3	Yes	HLR-WFR-A		WFR-E12	Yes
SPR-B7	CC-II	WHA-A4	Yes	WFR-A1	Yes		
SPR-B8	CC-I	WHA-A5	Yes	WFR-A2	Yes	HLR-WFR-F	
SPR-B9	CC-I	WHA-A6	Yes	WFR-A3	Yes	WFR-F1	Yes
SPR-B10	CC-I	WHA-A7	Yes	WFR-A4	Yes	WFR-F2	Yes
SPR-B11	CC-I	WHA-A8	Yes	WFR-A5	Yes		
SPR-B12	CC-I			WFR-A6	Yes	HLR-WFR-G	
SPR-B13	Yes	HLR-WHA-B		WFR-A7	Yes	WFR-G1	Yes
		WHA-B1	Yes	WFR-A8	Yes	WFR-G2	Yes
HLR-SPR-C		WHA-B2	Yes	WFR-A9	Yes		
SPR-C1	Yes	WHA-B3	Yes			HLR-WFR-H	
SPR-C2	Yes	WHA-B4	Yes	HLR-WFR-B		WFR-H1	CC-I
SPR-C3	Yes	WHA-B5	Yes	WFR-B1	Yes	WFR-H2	CC-II
SPR-C4	Yes	WHA-B6	Yes	WFR-B2	Yes	WFR-H3	Yes
SPR-C5	Yes			WFR-B3	Yes	WFR-H4	Yes
SPR-C6	Yes	HLR-WHA-C		WFR-B4	Yes		
		WHA-C1	Yes	WFR-B6	Yes	HLR-WFR-I	
HLR-SPR-D		WHA-C2	Yes	WFR-B7	Yes	WFR-I1	Yes
SPR-D1	Yes	WHA-C3	CC-II			WFR-I2	Yes
SPR-D2	CC-I	WHA-C4	Yes	HLR-WFR-C		WFR-I3	Yes
SPR-D3	CC-II	WHA-C5	Yes	WFR-C1	Yes		
SPR-D4	Yes	WHA-C6	Yes	WFR-C2	Yes	HLR-WPR-A	
SPR-D5	CC-II			WFR-C3	Yes	WPR-A1	Yes
		HLR-WHA-D		WFR-C4	Yes	WPR-A2	Yes
HLR-SPR-E		WHA-D1	Yes			WPR-A3	Yes
SPR-E1	Yes	WHA-D2	Yes	HLR-WFR-D		WPR-A4	Yes
SPR-E2	Yes			WFR-D1	Yes		
SPR-E3	Yes	HLR-WHA-E		WFR-D2	Yes	HLR-WPR-B	
SPR-E4	Yes	WHA-E1	Yes	WFR-D3	Yes	WPR-B1	Yes
SPR-E5	CC-II	WHA-E2	Yes	WFR-D4	Yes	WPR-B2	Yes
SPR-E6	Yes	WHA-E3	Yes	WFR-D5	Yes	WPR-B3	Yes
SPR-E7	Yes	WHA-E4	Yes	WFR-D6	Yes	WPR-B4	Yes
SPR-E8	Yes	WHA-E5	Yes			WPR-B5	Yes
				HLR-WFR-E		WPR-B6	CC-II
HLR-SPR-F		HLR-WHA-F		WFR-E1	Yes	WPR-B7	CC-I
SPR-F1	Yes	WHA-F1	Yes	WFR-E2	Yes		

Table A-3. Non-LWR CP Applications Based on the LMP Methodology: Applicability of ASME/ANS Non-LWR PRA Standard High-Level Requirements and Supporting Requirements to Additional PRA Elements (5 of 6)

WPR-B8	Yes	XFHA-A8	Yes	HLR-XFFR-A		HLR-XFPR-C	
WPR-B9	Yes	XFHA-A9	Yes	XFFR-A1	Yes	XFPR-C1	Yes
				XFFR-A2	Yes	XFPR-C2	Yes
HLR-WPR-C		HLR-XFHA-B		XFFR-A3	Yes	XFPR-C3	Yes
WPR-C1	Yes	XFHA-B1	Yes	XFFR-A4	Yes	XFPR-C4	Yes
WPR-C2	Yes	XFHA-B2	Yes	XFFR-A5	Yes	XFPR-C5	Yes
WPR-C3	Yes	XFHA-B3	Yes			XFPR-C6	CC-I
WPR-C4	Yes	XFHA-B4	Yes	HLR-XFFR-B		XFPR-C7	Yes
WPR-C5	Yes			XFFR-B1	Yes	XFPR-C8	Yes
		HLR-XFHA-C		XFFR-B3	Yes	XFPR-C9	Yes
HLR-WPR-D		XFHA-C1	Yes	XFFR-B4	Yes	XFPR-C10	Yes
WPR-D1	Yes	XFHA-C2	CC-II	XFFR-B5	Yes	XFPR-C11	CC-II
WPR-D2	Yes	XFHA-C3	CC-II			XFPR-C12	Yes
WPR-D3	CC-I	XFHA-C4	Yes	HLR-XFFR-C			
WPR-D4	CC-II	XFHA-C5	Yes	XFFR-C1	CC-I	HLR-XFPR-D	
WPR-D6	Yes	XFHA-C6	Yes	XFFR-C2	Yes	XFPR-D1	Yes
WPR-D7	Yes	XFHA-C7	Yes			XFPR-D2	Yes
WPR-D8	Yes	XFHA-C8	Yes	HLR-XFFR-D		XFPR-D3	Yes
WPR-D9	Yes	XFHA-C9	Yes	XFFR-D1	CC-I	XFPR-D4	Yes
WPR-D10	Yes	XFHA-C10	Yes	XFFR-D2	CC-II	XFPR-D5	Yes
WPR-D11	CC-I	XFHA-C11	Yes	XFFR-D3	Yes		
				XFFR-D4	Yes	HLR-XFPR-E	
HLR-WPR-E		HLR-XFHA-D				XFPR-E1	Yes
WPR-E1	Yes	XFHA-D1	CC-II	HLR-XFFR-E		XFPR-E2	CC-I
WPR-E2	Yes	XFHA-D2	Yes	XFFR-E1	Yes	XFPR-E3	CC-II
WPR-E3	Yes	XFHA-D3	CC-I	XFFR-E2	Yes	XFPR-E4	Yes
WPR-E4	Yes	XFHA-D4	Yes			XFPR-E5	Yes
WPR-E5	CC-II			HLR-XFFR-F		XFPR-E6	CC-I
WPR-E6	Yes	HLR-XFHA-E		XFFR-F1	Yes	XFPR-E7	Yes
WPR-E7	Yes	XFHA-E1	Yes	XFFR-F2	Yes	XFPR-E8	Yes
		XFHA-E2	Yes	XFFR-F3	Yes		
HLR-WPR-F		XFHA-E3	CC-II			HLR-XFPR-F	
WPR-F1	Yes	XFHA-E4	CC-II	HLR-XFPR-A		XFPR-F1	Yes
WPR-F2	Yes			XFPR-A1	Yes	XFPR-F2	Yes
WPR-F3	Yes	HLR-XFHA-F		XFPR-A2	Yes	XFPR-F3	Yes
		XFHA-F1	Yes	XFPR-A3	Yes	XFPR-F4	Yes
C.1.3.13 (XF)		XFHA-F2	Yes	XFPR-A4	Yes	XFPR-F5	CC-II
HLR-XFHA-A		XFHA-F3	Yes	XFPR-A5	Yes	XFPR-F6	Yes
XFHA-A1	Yes	XFHA-F4	Yes	XFPR-A6	Yes	XFPR-F7	Yes
XFHA-A2	Yes			XFPR-A7	CC-II		
XFHA-A3	Yes	HLR-XFHA-G				HLR-XFPR-G	
XFHA-A4	Yes	XFHA-G1	Yes	HLR-XFPR-B		XFPR-G1	Yes
XFHA-A5	Yes	XFHA-G2	Yes	XFPR-B1	Yes	XFPR-G2	Yes
XFHA-A6	Yes	XFHA-G3	Yes	XFPR-B2	Yes	XFPR-G3	Yes
XFHA-A7	Yes			XFPR-B3	Yes		

Table A-3. Non-LWR CP Applications Based on the LMP Methodology:

Applicability of ASME/ANS Non-LWR PRA Standard High-Level Requirements and Supporting Requirements to Additional PRA Elements (6 of 6)			
HLR-XFPR-H	HLR-OPR-B		
XFPR-H1 Yes	OPR-B1 Yes		
XFPR-H2 Yes	OPR-B2 Yes		
XFPR-H3 Yes	OPR-B3 Yes		
	OPR-B4 Yes		
C.1.3.14 (O)	OPR-B5 Yes		
HLR-OHA-A	OPR-B6 Yes		
OHA-A1 Yes	OPR-B7 Yes		
OHA-A2 Yes	OPR-B8 CC-II		
OHA-A3 CC-II	OPR-B9 CC-I		
OHA-A4 CC-I	OPR-B10 CC-I		
OHA-A5 Yes	OPR-B11 CC-I		
OHA-A6 Yes	OPR-B12 CC-I		
OHA-A7 Yes			
OHA-A8 CC-I	HLR-OPR-C		
OHA-A9 Yes	OPR-C1 Yes		
OHA-A10 Yes	OPR-C2 CC-I		
	OPR-C4 Yes		
HLR-OHA-B	OPR-C5 Yes		
OHA-B1 Yes	OPR-C6 CC-I		
OHA-B2 Yes	OPR-C7 Yes		
OHA-B3 Yes	OPR-C8 Yes		
HLR-OFR-A	HLR-OPR-D		
OFR-A1 CC-I	OPR-D1 Yes		
OFR-A2 Yes	OPR-D2 Yes		
OFR-A3 Yes	OPR-D3 Yes		
OFR-A4 CC-I	OPR-D4 Yes		
OFR-A5 Yes	OPR-D5 Yes		
OFR-A6 Yes	OPR-D6 CC-II		
OFR-A7 Yes	OPR-D7 Yes		
	OPR-D8 Yes		
HLR-OFR-B	OPR-D9 Yes		
OFR-B1 Yes			
OFR-B2 Yes	HLR-OPR-E		
OFR-B3 Yes	OPR-E1 Yes		
	OPR-E2 Yes		
HLR-OPR-A	OPR-E3 Yes		
OPR-A1 Yes	OPR-E4 Yes		
OPR-A2 Yes	OPR-E5 Yes		
OPR-A3 Yes			
OPR-A4 Yes			

Acronyms/Abbreviations

ADAMS	Agencywide Documents Access and Management System
ANS	American Nuclear Society
ASME	American Society of Mechanical Engineers
CC	Capability Category
CFR	*Code of Federal Regulations*
CP	construction permit
DBHL	design-basis hazard level
DBEHL	design-basis external hazard level
EAB	exclusion area boundary
ER	environmental report
EPZ	emergency planning zone
FR	*Federal Register*
LBE	licensing basis event
LMP	Licensing Modernization Project
NEI	Nuclear Energy Institute
non-LWR	non-light-water reactor
NRC	Nuclear Regulatory Commission
OL	operating license
POS	plant operating state
PRA	probabilistic risk assessment
PSAR	preliminary safety analysis report
QHO	quantitative health objective
RG	regulatory guide
RIM	reliability and integrity management
RSF	required safety function
SAMA	severe accident mitigation alternative
SAMDA	severe accident mitigation design alternative
SMR	small modular reactor
SR SSCs	safety-related systems, structures, and components

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A-6 American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants," New York, NY, 2021.

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Publications from the Nuclear Energy Institute (NEI) are available at their Web site: http://www.nei.org/ or by contacting the headquarters at Nuclear Energy Institute, 1776 I Street NW, Washington DC 20006-3708; telephone: 202-739-800; fax 202-785-4019.

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

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