U.S. NUCLEAR REGULATORY COMMISSION

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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) 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), manufacturing license (ML), standard design approval (SDA), 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, certifications, and approvals under 10 CFR Part 50 and 10 CFR Part 52 for such reactors. Upon conclusion of the

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This RG is being issued in draft form to involve the public in the development of regulatory guidance in this area. It has not received final staff review or approval and does not represent an NRC final staff position. Public comments are being solicited on this DG and its associated regulatory analysis. Comments should be accompanied by appropriate supporting data. Comments may be submitted through the Federal rulemaking Web site, http://www.regulations.gov, by searching for draft regulatory guide DG-1404. Alternatively, comments may be submitted to the Office of Administration, Mailstop: TWFN 7A-06M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, ATTN: Program Management, Announcements and Editing Staff. Comments must be submitted by the date indicated in the *Federal Register* notice.

Electronic copies of this DG, previous versions of DGs, and other recently issued guides are available through the NRC’s public Web site under the Regulatory Guides document collection of the NRC Library at https://www.nrc.gov/reading-rm/doc-collections/reg-guides/index.html. The DG is also available through the NRC’s Agencywide Documents Access and Management System (ADAMS) at http://www.nrc.gov/reading-rm/adams.html, under Accession No. ML22076A003. The regulatory analysis may be found in ADAMS under Accession No. ML22076A002.

rulemaking under way 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 early site permits, DCs, COLs, SDAs, and MLs 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.
  - 10 CFR 52.137, "Contents of applications; technical information," describes the information required to be included in FSARs supporting applications for SDAs.
  - 10 CFR 52.157, "Contents of applications; technical information in final safety analysis report," describes the information required to be included in FSARs supporting applications for MLs.

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.

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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.

- 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.

- 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; email: oira_submissions@omb.eop.gov.

Public Protection Notification

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

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B. DISCUSSION

Reason for Issuance

This RG provides 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, ML, SDA, 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 to identifying and describing the scope and level of detail for the fundamental safety functions of a design necessary for developing the safety analysis for the design. 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 LWR applicants 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, SDA, 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 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 the SAR that describe the fundamental safety functions of the design. TICAP provides guidance on the safety analysis necessary for an application consistent with the LMP approach as described in NEI 18-04, Revision 1, and endorsed by the NRC in RG 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.

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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 [DG-1404] 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.

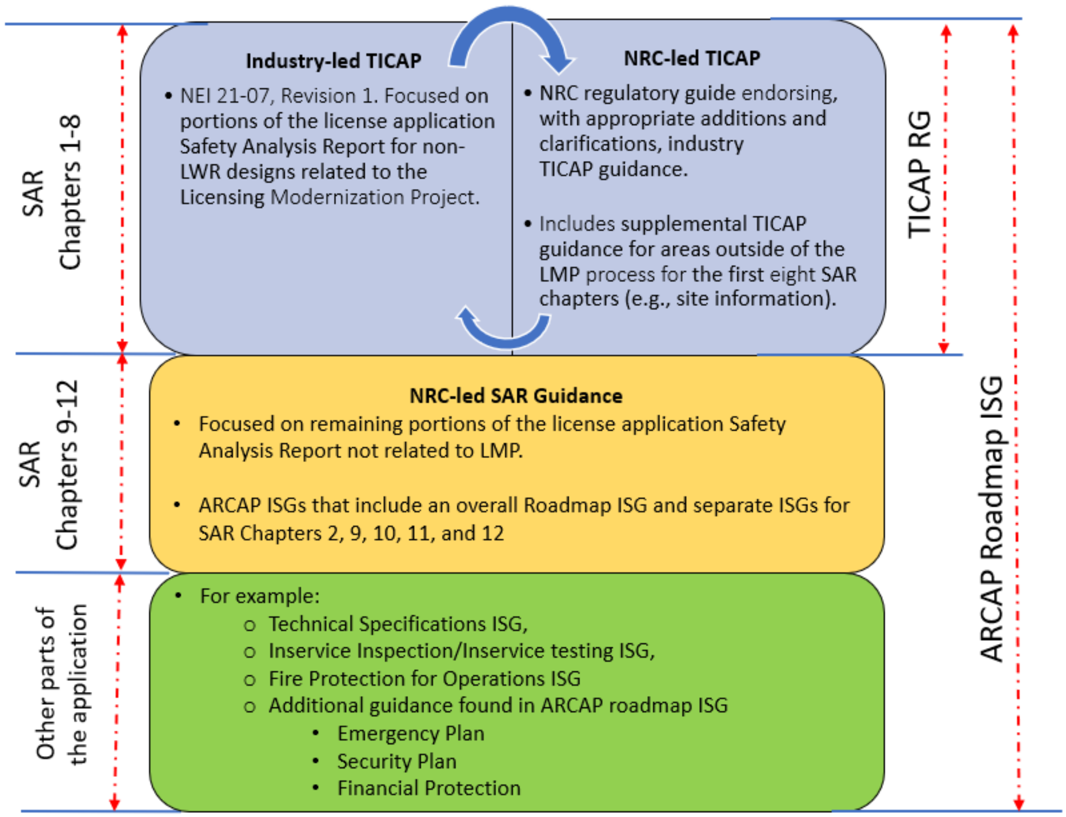

Figure 1. Relationship between ARCAP, TICAP, and the content of an application

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 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, ML, SDA, 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 preapplication 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.

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 italics.

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 a 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."[7]

b. Addition: 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,

7 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.

with sufficient justification. Appendix B to the ARCAP Roadmap ISG contains staff guidance on which regulations apply to non-LWRs.

c. Addition: NEI 21-07, Revision 1, provides an acceptable method for developing portions of the SAR for a CP application in accordance with 10 CFR Part 50 requirements. However, non-LWR applicants pursuing a CP under 10 CFR Part 50 using a risk-informed, performance-based approach other than the LMP should provide additional information than what is described in NEI 21-07, Revision 1. Specifically, 10 CFR 50.34(a) identifies the minimum technical information necessary in a CP application. Under 10 CFR 50.35(a), when the applicant has not supplied all of the technical information required to support the issuance of a CP that approves all proposed design features, the Commission may issue a CP provided that the Commission makes the findings identified in that section. The staff notes that the additional CP information described in this RG pertains to the first eight chapters of the SAR. The ARCAP roadmap ISG provides guidance for CP information outside the first eight chapters of the SAR.

d. 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.

e. Addition: In Chapter 1 of the SAR, in addition to the information identified in NEI 21-07, Revision 1, Section C.1, applicants should include summary tables with the following information, which appears in full elsewhere in the SAR:

   (1) The generic safety issues, unresolved safety issues, and Three Mile Island action items technically applicable to the design, and their proposed resolution (for generic safety issues, see NUREG-0933, Resolution of Generic Safety Issues (Ref. 21). The guidance on applicability of regulations in Appendix B to the ARCAP Roadmap ISG may provide useful insights in this area.

   (2) RGs directly applicable to the design, and whether the applicant proposes an alternative approach to satisfy a regulation rather than following the guidance in one of these RGs. If so, these should be discussed in the relevant portions of the SAR, including the technical justification for the alternate approach.

   (3) The consensus codes and standards (from ASME, the American Nuclear Society (ANS), the American Concrete Institute (ACI), the Institute of Electrical and Electronics Engineers, etc.) used in the design, and whether the applicant proposes to request an exemption from or alternative to the IEEE standard that is incorporated by reference into 10 CFR 50.55a. Regarding ASME, ANS, ACI, or other codes and standards used in the design, the applicant should also include a discussion of any deviations from these codes and standards and, as applicable, deviations from RGs in which the NRC staff endorsed the use of these codes and standards. In addition, codes and standards should also be discussed in the relevant portions of the SAR.

The guidance for providing these summary tables is consistent with previous NRC guidance for new reactors in RG 1.206, as well as the practice employed in FSARs for many operating plants. The staff finds these tables to be useful references for applicants preparing applications for NRC staff reviews, for licensees changing the licensing basis through applicable change processes (e.g., 10 CFR 50.59, "Changes, tests and experiments") or preparing license amendments to submit for NRC staff review.

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 a 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 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 a 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. 22), endorses with exceptions and clarifications the ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants," (Ref. 23), guidance for non-LWR PRAs and endorses with no exceptions and clarifications the NEI 20-09, "Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard" (Ref. 24), guidance on PRA peer reviews.

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. The NRC staff notes that additional guidance is being considered for development that would supplement the guidance in RG 1.247. Appendix A of this document identifies guidance that is being considered for development that could result in a revision of this Draft RG. 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).[8]

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 a 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 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 assumed in the radiological consequence evaluations. The following clarifications and additions are included:

a. Clarification: If the applicant has submitted separate licensing documents such as topical reports either during pre-licensing interactions or during application review with requests for staff review and approval, and if these separate documents are incorporated by reference in the SAR, then this may reduce the information that needs to be included in the SAR. Documents incorporated by reference in the SAR, which NEI 21-07, Revision 1, does not address, are considered part of the SAR and are therefore also subject to applicable change processes. Other types of documents submitted by applicants (e.g., white papers, information papers, regulatory engagement plan) that have not been formally reviewed and approved by the staff and are not incorporated by reference into the SAR will not reduce the information requirements for the SAR.

b. 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.

c. 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

[8] 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.

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). To conform to this guidance, the applicant has two options based on the outcome of the LMP approach:

(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" to demonstrate containment performance and to confirm that the EAB and LPZ doses are below the reference values in the regulations.

    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.

(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 LPZ.

d. 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." As noted above and in Appendix A, item 1, of this document, the staff is developing guidance related to PRA that, if approved, would provide supplemental guidance for PRA preparation at the CP stage.

e. Clarification: Section C.2.1.1 of NEI 21-07 states that “At the CP stage, neither the plant design nor the PRA is expected to have the level of maturity that will be necessary to support an OL application. At the CP application stage, the applicant should describe its ultimate intended approach for qualifying the PRA. If conformance to ASME/ANS RA-S-1.4-2021 is planned, a simple statement to that effect should be sufficient.” The “simple statement” is only regarding the commitment to conform to ASME/ANS RA-S-1.4-2021 at the OL stage. The description or a summary of PRA in a CP application, however, is broader than a simple commitment to the standard. As noted in 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.d 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.

f. Clarification: Section C.2.1.1 of NEI 21-07 states that “Note: This guidance document does not address SAR content for a PRA that has not been peer reviewed using the non-LWR PRA standard. In such an instance, the information to be provided on the PRA, either in the SAR or other documentation, may be more extensive than the guidance provided herein.” NEI 21-07, Revision 1, however, addresses the SAR content for a CP application for which no peer review has been performed; accordingly, this Note should be applied only to SAR content for applications other than a CP application.

g. Addition: 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 additional 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 additional information is (1) to demonstrate compliance with 10 CFR Part 100, “Reactor Site Criteria” (Ref. 25), 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) SR SSCs must be protected from or designed to withstand the corresponding DBHL with no adverse impact on their ability to perform their required safety functions (RSFs),

   (2) 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

   (3) 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.

   Determination of these special treatments for NSRST SSC will be made by the integrated design process panel (IDPP) in accordance with NEI 18-04 and RG 1.233. The draft ARCAP ISG, DANU-ISG-2022-02, “Site Information,” (Ref. 26), contains additional guidance on one acceptable approach to determining the scope and level of detail of the site information to be provided.

h. 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 non-PRA analysis and calculation methodologies used to establish their licensing bases. A change to any of the evaluation methodologies used in the licensing basis is one of the criteria in the existing facility change process (e.g., 10 CFR 50.59); therefore, for the facility change process to be effective, the licensing basis needs to describe these evaluation methodologies.

4. Licensing-Basis Events

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 a SAR. After identifying the LBEs, Chapter 3 of a 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, DBAs, and BDBEs in Chapter 3 of the SAR should include a description of the models, site characteristics, and 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). Other additions/clarifications related to this topic include:

   (1) The supporting data should include the data that is significant to determining whether the frequency-consequence targets and quantitative health objectives (QHOs) are met and the development of the analysis conclusions on risk significance, SSC classification, or DID

adequacy.

(2) For Section C.3.3.1 of NEI 21-07, Revision 1, the staff clarifies that for any AOO with a release, all of the information in the second bulleted list on NEI 21-07 page 33 should be provided in addition to the information in the first paragraph on that page.

(3) For Section C.3.4.1 of NEI 21-07, Revision 1, the staff clarifies that for the most limiting DBE used to map into each DBA, all of the information in the second bulleted list beginning on NEI 21-07 page 34 should be provided in addition to the information in the first bulleted list on page 34.

(4) For Section C.3.5.1 of NEI 21-07, Revision 1, the staff clarifies that for any high-consequence BDBEs and other BDBEs that bound the risks associated with the collection of BDBEs, all of the information in the second bulleted list on NEI 21-07 page 36 should be provided in addition to the information in the first bulleted list on page 35.

b. Clarification: The second-to-last paragraph in each of Sections C.3.3.1, C.3.4.1, and C.3.5.1 of NEI 21-07, Revision 1, appears to conflict with guidance in Section C.2.1.1 on the level of detail in the SAR for non-DBA LBEs. Therefore, the staff provides the following clarification: Section C.2.1.1 of NEI 21-07, Revision 1, contains adequate guidance on the level of detail in the SAR to describe non-DBA LBEs. If there is event-specific information associated with the radiological consequence evaluation, the applicant may elect to provide that information in Chapter 3 of the SAR instead of in Section 2.1.1. Further details on the models, site characteristics, and supporting data associated with the calculation of probabilities, mechanistic source terms, and radiological consequences, beyond the content specified for Section 2.1.1 of the SAR (or Chapter 3 for event-specific information), are part of the PRA documentation and can be included in the plant records.

c. Addition: In addition to the material identified in NEI 21-07, Revision 1, Section C.3, Chapter 3 of the SAR should also discuss the following:

(1) Aircraft impact assessments (10 CFR 50.150, "Aircraft impact assessment" (Ref. 27)). The aircraft impact rule requires nuclear power plant designers to rigorously assess their designs for features and functional capabilities that could provide additional inherent protection to withstand the effects of an aircraft impact. New nuclear power reactor facilities will be inherently more robust with regard to aircraft impact than facilities designed in the absence of the aircraft impact rule. When implementing 10 CFR 50.150, applicants should use RG 1.217, "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts" (Ref. 28), which describes a method that the NRC staff considers acceptable for use in satisfying the regulations at 10 CFR 50.150. RG 1.217 endorses the industry guidance document NEI 07-13, Revision 8P, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," issued April 2011 (Ref. 29). The ARCAP Roadmap ISG contains guidance for NRC staff on aircraft impact assessments and may be informative for applicants.

(2) Mitigation of Beyond-Design-Basis Events (MBDBE) (10 CFR 50.155, "Mitigation of beyond-design-basis events" (Ref. 30)): 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. 31), 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. 32), 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. 33), 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. 34), 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. 35), 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. 36), provides guidance on how to develop the application content for demonstrating that the requirements of 10 CFR 50.155(b)(2) are met.[9]

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 a SAR following NEI 21-07, Revision 1, should provide the overall plant risk performance summary for the proposed

9 SRP Section 19.4, "Strategies and Guidance to Address Loss-of-Large Areas of the Plant Due to Explosions and Fires" (Ref.41) 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.

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. 37).

- 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 a 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:

a. Addition: 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 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. Alternatively, the applicant should ensure that its DID process involves incorporating DID into design features, operating and emergency procedures, and other programmatic elements to ensure that performance requirements are maintained throughout the life of the plant. An applicant that chooses not to use the approach endorsed in RG 1.233 will need to explain its approach to DID and describe how it addresses DID in the application.

b. Addition: For each of the three plant performance metrics discussed above in Section 5 of this document (section C.4 of the application guidance), in addition to the results and margins, the SAR Chapter 4 should address the following (where different from the analysis performed for Chapter 3):

   (1) postulated site parameters or site characteristics (e.g., meteorology, off-site population distribution, EAB size) used in the analysis,

   (2) postulated locations of individual members of the public,

   (3) source of dose (cloud shine, inhalation, ground shine),

(4) analysis method used,

(5) key modeling assumptions (e.g., initial and boundary conditions, emergency preparedness measures, source terms, timing and duration of release, credit for medical treatment, early and latent fatality risk coefficients) used in the analysis,

(6) modes of operation (full power, low power, shutdown, and refueling) considered in the analysis,

(7) consideration of multiple units and other radiological sources on the site, and

(8) uncertainty/sensitivity analysis performed.

c. Addition: Additional information for human factors considerations: If not included in SAR Chapter 6 or 7 an applicant should also address human factors considerations such as operating experience review, safety function review, human action task analysis, human-system interface design, procedures, training, verification and validation, and human performance monitoring. Guidance on ensuring a holistic approach to the human factors engineering program appears in the draft ARCAP ISG, DANU-ISG-2022-05, "Organization and Human-System Considerations," (Ref. 38).

d. Addition: In Chapter 4 of the SAR, the applicant should discuss how changes to the design are assessed for possible effects on the DID analysis. In particular as described in NEI 18-04, Revision 1, and RG 1.233, in describing the change control process, the applicant should address how it will reevaluate the baseline DID evaluation results to determine which programmatic or plant capability attributes in each layer of defense would be affected by proposed changes. Changes that affect the definition or evaluation of LBEs, the safety classification of SSCs, or the risk significance of LBEs or SSCs should be assessed.

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 a SAR following NEI 21-07, Revision 1, should describe the applicant's approach to designating SSC safety functions and classifications in accordance with the PRA safety functions (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 and additional information.

(1) Addition: The requirements in 10 CFR 50.34(a)(3), 52.79, 52.137, and 52.157 to propose PDC includes a requirement, for both LWRs and non-LWRs, to establish the necessary design,

fabrication, construction, testing, and performance requirements for SSCs important to safety (as described in paragraph C.6.a.(2) of this staff position below). As provided in Appendix A to Part 50, the GDC are intended to provide guidance in establishing the PDC for nuclear power plants such as non-LWRs. Applicants addressing less than the full scope of PDC must request an exemption from the applicable requirements for providing proposed PDCs and provide suitable justification for the exemption. For example, the justification may be that, to address specific elements of PDC scope, the applicant has complied with other regulatory requirements that compel the applicant to provide the relevant information in other portions of the application. 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 (ARDC) 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. The NRC staff considers the proposed PDC-RFDC and PDC-CDC to be equivalent in establishing the necessary requirements for SSCs that provide reasonable assurance that the facility can be operated without undue risk to public health and safety. The staff therefore considers the two-tiered approach to be an acceptable method for proposing PDC for non-LWRs using the LMP methodology.

(3) Addition: 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 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 fuel qualification: In addition to the material identified in NEI 21-07, Revision 1, Section C.5, Chapter 5 of a 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, other information such as fuel design limits, is attributed to or identified with fuel performance and its qualification. In particular, fuel cannot be qualified without irradiation data collected over certain time frames. Accordingly, non-LWR applicants may use existing data (e.g., Advanced Gas-Cooled Reactor program data, legacy metal fuel data), to some degree, to support regulatory licensing. Two documents provide additional background on non-LWR fuel qualification: (1) NRC guidance in NUREG-2246, "Fuel Qualification for Advanced Reactors," issued March 2022 (Ref. 39), and (2) an example of a generic fuel qualification topical report and associated safety evaluation applicable to multiple non-LWR designs, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance," issued December 2020 (Ref. 40). The applicant's discussion of fuel qualification 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. Sufficient information should be available to support findings of the following:

(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 a 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 application should include a discussion in the SAR on how the applicant intends to confirm, at the OL stage, that the reliability and capability performance targets assumed in the PRA have been met. The application should describe any testing and validation planned to confirm SR SSC performance capabilities and availability, including any special treatment to be applied to the SR SSCs.

The safety-related design criteria[10] are derived from the RFDC, which are developed from the RSFs determined in the LBE selection process described in Chapters 2 and 3 of NEI 21-07, Revision 1.

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10 For SSCs classified as SR, the design criteria are referred to as Safety-Related Design Criteria (SRDC). These are derived from the Required Functional Design Criteria (RFDC) that are in turn developed from the Required Safety Functions (RSFs) determined in the LBE selection process as discussed in NEI 18-04, Section 3. RSFs are those safety functions that must be fulfilled to keep the DBEs within the F-C Target. RFDCs are taken down to a lower level and form a transition to SSC-level criteria. RFDCs are defined to capture design-specific criteria that may be used to supplement or modify the applicable GDCs or ARDCs in the formulation of PDCs. RSFs and RFDCs are technology- and design-specific and are framed at the function level. After SR SSCs have been selected to perform the RSFs, the SRDCs are defined at the SSC level in a manner that assures meeting the RFDCs and the RSFs for the specific SSC selected to perform the RSFs.

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. 41), 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 a 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.

Chapter 6 also establishes the DBHLs associated with NSRST SSCs and SSCs that are non-safety related with no special treatment (NST). The design requirements for NSRST and NST SSCs are determined by the need to protect SR SSCs in the performance of their RSFs from adverse effects from the failure of NSRST or NST SSCs during and after DBEs.

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. Clarification and additional information: In addition to describing the DBHLs as stated in NEI 21-07, Revision 1, Section C.6, on SAR Chapter 6, the application should also include the translation of DBHLs to loads on SSCs, evaluation of those loads, and related design analyses. NEI 21-07, Revision 1, does not provide guidance in this area; applicants can refer to Section C.I.3 of RG 1.206, Revision 0 (Ref. 42) for guidance.[11] For an advanced non-LWR application, this material may be included in Chapter 2 of the SAR or in reports submitted separately from but incorporated by reference in the SAR. The scope and level of detail of these calculations are design and site specific. Preapplication interaction with the staff may be appropriate to determine the necessary level of information.

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11 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.

b. Addition: In addition to the material identified in NEI 21-07, Revision 1, preparation of Chapter 6 of the SAR should also include the following:

(1) If the facility includes instrumentation and control systems classified as SR, the applicant should describe the special treatments applied to them and their components and analyze their capability to perform their credited safety functions.[12]

(2) The applicant should justify the use of the chosen code or standard for the particular reactor described in the application. If the facility includes SR equipment and components designed with materials that will be called upon to withstand high-temperature service conditions, the applicant may reference ASME Boiler and Pressure Vessel Code, Section III, Division 5, "High Temperature Reactors," (Ref. 44), as endorsed by Regulatory Guide 1.87, "Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors,'" Revision 2 (Ref. 45). Appendix A of this document provides a listing of draft documents under development that the staff is considering that could potentially supplement the guidance found in RG 1.87.

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 a 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. For those NSRST SSCs whose reliability and capabilities have not been confirmed at the CP stage, the application should include a discussion in the SAR on how the applicant intends to confirm, at the OL stage, that reliability and capability performance targets have been met. The application should describe any testing and validation planned to confirm NRST SSC performance capabilities and availability, including any special treatment to be applied to the NSRST SSCs.

As discussed earlier, Chapter 6 of the SAR establishes DBHL requirements and identifies design parameters for NSRST and NST SSCs.

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.

11 "Design Review Guide (DRG): Instrumentation and Controls for Non-Light Water Reactor (Non LWR) Reviews," dated February 26, 2021 (Ref. 48) provides guidance to the NRC staff for review of this topic. The DRG 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. Addition: In addition to the material identified in NEI 21-07, Revision 1, Section C.7, preparation of Chapter 7 of the SAR should also consider the following:

(1) If the facility includes instrumentation and control systems identified as NSRST, the applicant should describe the special treatments applied to the I&C systems and components and analyze their capability to perform their credited functions.[13]

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 a 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 a discussion in the SAR to develop 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.

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.

13 "Design Review Guide (DRG): Instrumentation and Controls for Non-Light Water Reactor (Non LWR) Reviews," dated February 26, 2021, provides guidance to the NRC staff for review of this topic. The DRG 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.

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. 46), 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
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*

REFERENCES [14]

1. Nuclear Energy Institute (NEI), 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,” Washington, DC, February 2022. (Agencywide Documents Access and Management System (ADAMS) Accession No. ML22060A190)

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.”

4. NRC, “Policy Statement on the Regulation of Advanced Reactors,” *Federal Register*, Vol. 73, No. 199, October 14, 2008, pp. 60612–60616 (73 FR 60612).

5. NRC, RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” Washington, DC.

6. NRC, RG 1.181, “Content of the Updated Final Safety Analysis Report in Accordance with 10 CFR 50.71(e),” Washington, DC.

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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14 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.

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 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)

12. NRC, “NRC Non-Light Water Reactor Near-Term Implementation Action Plans,” July 2017. (ML17165A069)

13. NRC, “NRC Non-Light Water Reactor Mid-Term and Long-Term Implementation Action Plans,” July 2017. (ML17164A173)

14. NRC, 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,” August 31, 2010. (ML1025210405)

15. NRC, SRM-SECY-11-0024, “Staff Requirements—SECY-11-0024—Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews,” May 11, 2011. (ML111320551)

16. NRC, DANU-ISG-2022-01, “Review of Risk-Informed, Technology-Inclusive Advanced Reactor Applications-Roadmap,” Washington, DC, April 2022. (ML22048B546)

17. NRC, NUREG 0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” Washington, DC

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

19. NRC, Management Directive (MD) 6.6, “Regulatory Guides,” Washington, DC.

20. International Atomic Energy Agency (IAEA) Specific Safety Requirements (SSR), No. SSR-2/1, “Safety of Nuclear Power Plants: Design,” Vienna, Austria, 2016.

21 NRC, NUREG-0933, Resolution of Generic Safety Issues (Formerly entitled "A Prioritization of Generic Safety Issues") (NUREG-0933, Main Report with Supplements 1–35), available at https://www.nrc.gov/sr0933/index.html

22. NRC, RG 1.247 for trial use, “Acceptability of Probabilistic Risk Assessment Results for Non-Light-Water Reactor Risk-Informed Activities,” Washington, DC.

23. 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.

24. NEI 20-09, “Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard,” Washington DC, May 2021. (ML21125A284)

25. 10 CFR Part 100, “Reactor Site Criteria.”

26. NRC, DANU-ISG-2022-02, “Site Information,” Washington, DC. (ML22048B541)

27. 10 CFR 50.150, "Aircraft impact assessment."

28. NRC, RG 1.217, Revision 0, "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts," Washington, DC, August 2011. (ML092900004)

29. NEI 07-13, Revision 8P, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Washington, DC, April 2011. (ML111440006)

30. 10 CFR 50.155, "Mitigation of beyond design-basis events."

31. NRC, RG 1.226, Revision 0, "Flexible Mitigation Strategies for Beyond-Design-Basis Events," Washington, DC, June 2019. (ML19058A012)

32. NEI 12-06, Revision 4, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Washington, DC, December 2016. (ML16354B421)

33. NRC, RG 1.227, Revision 0, "Wide-Range Spent Fuel Pool Level Instrumentation," Washington, DC, June 2019. (ML19058A013)

34. 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,'" Washington, DC, August 2012. (ML12240A307)

35. NRC, Final Rule, "Mitigation of Beyond-Design-Basis Events," *Federal Register*, Vol. 84, No. 154 August 9, 2019, pp. 39684–39722 (84 FR 39684).

36. NEI 06-12, Revision 2, "B.5.b Phase 2 & 3 Submittal Guideline," Washington DC, December 2006.

37. 10 CFR Part 20, "Standards for Protection against Radiation."

38. NRC, DANU-ISG-2022-05, "Organization and Human System Considerations," Washington, DC, April 2022. (ML22048B542)

39. NRC, NUREG-2246, "Fuel Qualification for Advanced Reactors, Draft Report for Comment," Washington, DC, March 2022. (ML22063A131)

40. EPRI, EPRI-AR-1(NP)-A, "Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO)-Coated Particle Fuel Performance,", December 2020. (ML20336A052)

41. NRC, RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," Washington, DC, May 2006. (ML061090627)

42. NRC, RG 1.206, Revision 0, "Combined License Applications for Nuclear Power Plants (LWR Edition)," Washington DC, June 2007. (ML070720184)

43. NRC, Design Review Guide (DRG): Instrumentation and Controls for Non-Light Water Reactor (Non LWR) Reviews," Washington DC, February 26, 2021.

44. ASME Boiler and Pressure Vessel Code, Section III, Division 5, "High Temperature Reactors," New York, NY 2021

45 NRC, RG 1.87, Revision 2, "Acceptability of ASME Code Section III, Division 5, 'High Temperature Reactors,'" Washington, DC, January 2023. (ML22101A263)

46. NRC, MD 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," Washington, D.C.

APPENDIX A

DRAFT TECHNOLOGY INCLUSIVE CONTENT OF APPLICATION PROJECT (TICAP) GUIDANCE DOCUMENTS UNDER DEVELOPMENT AS OF MAY 2023

The purpose of this appendix is to provide a list of draft guidance documents that are under consideration for future updates to this TICAP draft regulatory guide (DG) (i.e., DG-1404). These draft documents are under development and have not received a complete staff review; therefore, they do not represent official NRC staff positions. If an applicant relies on any one of these draft documents, the applicant will be at risk that a final NRC position will conflict with the position provided in the draft document. The table below lists the guidance under development that has the potential to cause the TICAP DG to be updated to reflect the final versions of the draft documents listed in the second column.

Item #	Draft Document Being Considered for Possible Update	Comments
1	An appendix to this document is being considered for development to provide additional guidance for the scope, level of detail, elements, and plant representation for a probabilistic risk assessment (PRA) supporting a Licensing Modernization Project (LMP)-based construction permit application	This guidance, if issued, would supplement the guidance in RG 1.247 for trial use. During an April 18, 2023, public meeting on the topic (see: https://www.nrc.gov/pmns/mtg?do=details&Code=20230362) the NRC staff outlined an approach for the development of guidance in this area.
2	Draft interim staff guidance (ISG) is being considered for development associated with the relationship between the type of licensing applications and the Capability Categories of the supporting requirements in ASME/ANS RA-S-1.4-2021, "Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plant," (i.e., NLWR PRA standard).	This guidance, if issued, would supplement the guidance found in RG 1.247 for trial use. The guidance is being considered because some supporting requirements in the NLWR PRA standard are not applicable to certain plant applications or stages, while other supporting requirements need some clarification to understand how they can be achieved.
3	Draft Guide-1413 (proposed RG 1.254), "Technology-Inclusive Identification of Licensing Events for Commercial Nuclear Plants," (ADAMS Accession No. ML22272A042)	This guidance, if issued, would supplement the guidance found in RG 1.247 for trial use. The guidance would provide the staff's technology-inclusive guidance for identifying initiating events, delineating event sequences and licensing events that can be used to inform the design basis, licensing basis, and content of applications for commercial nuclear plants. Several of the beginning steps proposed in this guidance are applicable to the development of a probabilistic risk assessment (PRA)

Item #	Draft Document Being Considered for Possible Update	Comments
4	A Draft ISG is being considered for development that would provide guidance for treatment of consequence uncertainty in a PRA.	The guidance, if issued, would supplement the guidance found in RG 1.247 for trial use. Key to the approach in RG 1.247 for trial use is the development of frequency consequence criteria. While guidance for the treatment of uncertainty for the frequency of an event is considered sufficient, the staff is considering the development of additional guidance for the treatment of uncertainty in consequence evaluations.
5	DANU-ISG-2023-01, "Material Compatibility"	The guidance DANU-ISG-2023-01, "Material Compatibility" would identify areas of review that could be necessary in a submittal seeking to use materials that would be allowed under American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, "Rule for the Construction of Nuclear Facility Components," Division 5, "High Temperature Reactors" (Section III-5) (ASME, 2017). Section III-5 would specify the mechanical properties and allowable stresses to be used for design of components in high temperature reactors (HTRs). However, as stated in Section III-5, HBB 1110(g), the rules do not provide methods to evaluate deterioration that may occur in service as a result of corrosion, mass transfer phenomena, radiation effects, or other material instabilities. This ISG would identify information that should be included in non-LWR applications to satisfy applicable design requirements including qualification and monitoring programs for safety-significant structures, systems, and components (SSCs).