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Sent: Friday, May 14, 2021 11:48 AM
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Subject: Draft White Papers Associated with the Development of a TICAP Regulatory Guide and ARCAP Roadmap Interim Staff Guidance
Attachments: ARCAP ISG - Roadmap.docx; TICAP - Draft Regulatory Guide DRG R2.docx

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Southern Company Services
Licensing and Policy Director – Next Generation Reactors

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The purpose of this email is to transmit to you the attached two draft white papers containing a preliminary proposed structure for the following documents:

- Technology Inclusive Content of Application Project (TICAP) Regulatory Guide
- Advanced Reactor Content of Application Project (ARCAP) Roadmap Interim Staff Guidance (ISG) Document

These draft white papers are being issued in parallel with the NRC staff's review of the April 15, 2021 draft TICAP industry guidance document found in ADAMS at Accession No. ML21106A013. The main purpose of providing these draft white papers to you at this early stage of their development is to engage stakeholders on the NRC staff's initial high-level considerations on issues to be considered in such guidance documents. With the draft white papers being at a very early stage, their specific content is very preliminary and is going to change. The intended goal of the TICAP RG is to endorse the industry-developed TICAP guidance document and the ARCAP Roadmap ISG is intended to identify key guidance documents that will support the proposed Part 53 rulemaking. The combination of these documents is intended to provide guidance for advanced reactor applicants in the near term and we understand that additional revisions of the guidance may be necessary in the future to address a broader set of applicants. To this end we expect to discuss how these draft white papers might further inform the development of industry’s TICAP guidance document at the upcoming TICAP workshops scheduled for May 19th and May 26th.

This email (including the attachments) will be made publicly available in ADAMS such that the documents can be referenced in the upcoming TICAP workshops.

Please let me know if you have any questions.

Sincerely,

Joe Sebrosky
Senior Project Manager
Advanced Reactor Policy Branch
Office of Nuclear Reactor Regulation
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Mail Envelope Properties (MN2PR09MB5385236050D30798D6D00BB5F8509)

Subject: Draft White Papers Associated with the Development of a TICAP Regulatory Guide and ARCAP Roadmap Interim Staff Guidance
Sent Date: 5/14/2021 11:47:54 AM
Received Date: 5/14/2021 11:47:00 AM
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Files	**Size**	**Date & Time**
MESSAGE	2289	5/14/2021 11:47:00 AM
ARCAP ISG - Roadmap.docx	392943	
TICAP - Draft Regulatory Guide DRG R2.docx	391404	

Options

Priority:	Normal
Return Notification:	No
Reply Requested:	No
Sensitivity:	Normal
Expiration Date:	

This draft staff white paper has been prepared and is being released to support ongoing public discussions. This draft white paper uses an interim staff guidance (ISG) format because the staff is considering using this format to provide staff guidance in the near future to support the review of advanced reactor applications. This draft white paper is being issued in parallel with the NRC staff's review of draft industry guidance. The main purpose of this document at this early stage of advanced reactor guidance development is to engage stakeholders on the staff's initial high-level considerations on issues to be considered in such guidance.

This paper has not been subject to NRC management and legal reviews and approvals, and its contents are subject to change and should not be interpreted as official agency positions.

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

DANU [XX]-ISG-[YYYY-##]

"Guidance for Performing the Review of a Technology-Inclusive Advanced Reactor Application - Review Roadmap"

Interim Staff Guidance

May X, 2021

DANU [XX]-ISG-[YYYY-##]
"Guidance for Performing the Review of a Technology-Inclusive Advanced Reactor Application-Review Roadmap"
Interim Staff Guidance

ADAMS Accession No.: MLxxxxxxxxx TAC: xxxxxx

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INTERIM STAFF GUIDANCE

"GUIDANCE FOR PERFORMING THE REVIEW OF A

TECHNOLOGY-INCLUSIVE

ADVANCED REACTOR APPLICATION -

REVIEW ROADMAP"

DANU-ISG-YYYY-##

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC, or Commission) staff is providing this interim staff guidance (ISG) to facilitate the review of anticipated advanced reactor applications for an advanced reactor construction permit (CP), operating license (OL), combined license (COL), manufacturing license (ML), standard design approval (SDA), or design certification (DC) under Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. X), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. X). It is anticipated that this guidance will be updated to support the technology-inclusive, risk-informed and performance-based advanced nuclear reactor license and permit applications submitted under 10 CFR) Part 53, "Licensing and Regulation of Advanced Nuclear Reactors," once that regulation is final.

This guidance found in this ISG provides a general overview of the information expected to be included in an advanced reactor application, and a review roadmap for NRC staff with the principal purpose of ensuring consistency, quality and uniformity of staff reviews, and to present a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. While specific sections of the information described in this ISG are primarily aligned with the Licensing Modernization Project (LMP) methodology as one acceptable process for use when developing portions of an application, the concepts and general information may be used to inform the review of an application submitted using other methodologies (as applicable) such as a maximum hypothetical accident, or deterministic approaches. Other sections of the information described in this ISG are generally applicable and independent of the methodology used to develop an advanced reactor application.

It is also a purpose of this ISG to make information about regulatory matters widely available and to improve communication and understanding of the staff review process by interested members of the public and the nuclear power industry by providing a roadmap for developing all portions of an application.

BACKGROUND

As the NRC prepares to review and regulate a new generation of advanced reactors, the staff has previously recognized the need to establish, and the benefits of having, a flexible regulatory framework. In this ISG, the term "advanced reactor" is used in the context of the Nuclear Energy Innovation and Modernization Act (NEIMA). NEIMA included a definition for "advanced nuclear reactor" that was further refined by the Energy Act of 2020. The definition of advanced nuclear reactor found in the Energy Act of 2020 includes:

> (1) ADVANCED NUCLEAR REACTOR. —The term 'advanced nuclear reactor' means—
> (A) a nuclear fission reactor, including a prototype plant (as defined in sections 50.2 and 52.1 of title 10, Code of Federal Regulations (or successor regulations)), with significant improvements compared to reactors operating on the date of enactment of the Energy Act of 2020, including improvements such as—
> (i) additional inherent safety features;
> (ii) lower waste yields;
> (iii) improved fuel and material performance;
> (iv) increased tolerance to loss of fuel cooling;
> (v) enhanced reliability or improved resilience;
> (vi) increased proliferation resistance;
> (vii) increased thermal efficiency;
> (viii) reduced consumption of cooling water and other environmental impacts;
> (ix) the ability to integrate into electric applications and nonelectric applications;
> (x) modular sizes to allow for deployment that corresponds with the demand for electricity or process heat; and
> (xi) operational flexibility to respond to changes in demand for electricity or process heat and to complement integration with intermittent renewable energy or energy storage; and
> (B) a fusion reactor.

In SECY 20-0032, "Rulemaking Plan On "Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (Rin-3150-Ak31; Nrc-2019-0062)" (Ref. x), the staff further clarified its interpretation of the advanced reactors described in NEIMA to include light-water small modular reactors (SMRs), non-light-water reactors (non-LWRs), and fusion reactor designs.

The NRC described 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," (Ref. x). The staff then developed "NRC Non-Light Water Reactor Near Term Implementation Action Plans" (Ref. x), and "NRC Non-Light Water Reactor Mid-Term and Long-Term Implementation Action Plans" (Ref. x), to identify specific activities that the NRC will conduct in the near-term, mid-term, and long term timeframes. Similarly, the Commission encouraged the use of a performance based technology inclusive licensing framework for SMRs in SRM-COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance Safety Focus of Small Modular Reactor Reviews," and SRM-SECY-11-0024, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews."

To ensure review readiness to regulate a new generation of advanced reactors, a key element of this new and flexible regulatory framework is to standardize the process for the development of content within an advanced reactor application to promote uniformity among applicants. A standardized process for the development of the content of application for advanced reactors also ensures review consistency and predictability from NRC staff, and presents a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. The development of applications for NRC licenses, permits, certifications, and approvals is a major undertaking, in that an applicant must provide sufficient information to support the agency's safety findings. The needed information and level of detail will vary according to the design and whether an application is for a construction permit, design approval, design certification, operating license, combined license, or other action.

The NRC staff has had success with a standard content of application methodology for large-LWRs. The NRC's efforts to standardize the format and content of applications for LWRs are reflected in RG 1.70, issued in the 1970s, and RG 1.206, issued in 2007 and revised in 2018. Guidance documents, such as NUREG 0800 and numerous other documents on specific technical areas, address the suggested scope and level of detail for applications. While it is not the intent of this ISG to re-create a NUREG-0800 type broad spectrum of review guidance for advanced reactors, it is the staff's intention to leverage the previous experience and insights gained from having the benefit of standard application content principles in this ISG.

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

- The Advanced Reactor Content of Application Project (ARCAP), and
- The Technology-Inclusive Content of Application Project (TICAP).

The ARCAP is an NRC-led activity and is intended to provide guidance for a complete advanced reactor application that supports either10 CFR Part 50 or 10 CFR Part 52, and the ongoing 10 CFR Part 53 rulemaking effort. As a result, ARCAP is broad and encompasses several industry-led, and NRC-led guidance documents aimed at ensuring a consistent approach to the development of each application document. A complete advanced reactor application is expected to include, among other things, a SAR, a Quality Assurance plan, a Fire Protection program, Emergency and Physical Security plans, etc. The information described in this ISG summarizes the results of the NRC led ARCAP efforts.

The TICAP is an industry-led activity, and is focused on providing guidance on the appropriate scope and depth of information related to the specific portions of the SAR that describe the fundamental safety functions of the design, and details the affirmative safety case[1] for each applicant consistent with the LMP approach. TICAP's focus on those measures needed to address risks posed by non-LWR and SMR technologies will help an applicant provide sufficient information on the design and programmatic controls, while avoiding an excessive level of detail on less important parts of a plant. The specific portions of the SAR applicable to the scope of NEI 21-xx are described below in more detail. Based on the limited scope of the TICAP guidance, TICAP's scope is encompassed by and supplemented by the ARCAP guidance. The

[1] The applicant or the licensee is still responsible for demonstrating compliance with all applicable regulations and may request exemptions as appropriate.

ARCAP will describe the guidance for the specific areas of the SAR that are outside the scope of the LMP process (i.e., not covered by TICAP) such as site information, and information consistent with use of the American Society of Mechanical Engineers (ASME) Section III, Division 5 construction codes.

Figure 1 below illustrates the nexus between ARCAP, TICAP, and other guidance in relation to an advanced reactor application.

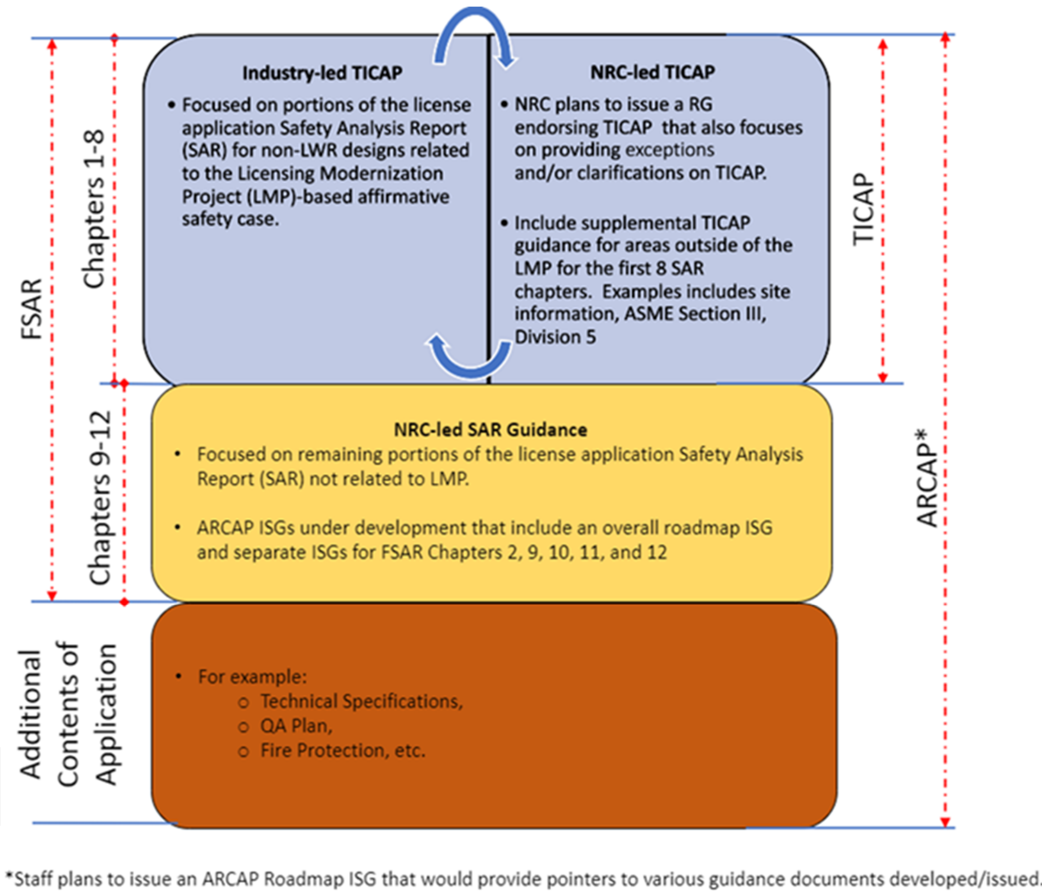

Figure 1: Nexus between ARCAP, TICAP, and the content of an application.

The LMP process is described in Nuclear Energy Institute (NEI) document NEI–18–04, Revision 1, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development," and endorsed by the NRC via Regulatory Guide (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." The LMP methodology outlines a risk-informed, and performance-based approach for use by reactor developers to identify and select licensing basis events (LBEs) applicable to the site under consideration, classify systems, structures, and components (SSCs), determine special treatments and programmatic controls, and assess the adequacy of a design in terms of providing layers of defense-in-depth (DID). In addition, the

LMP methodology and RG 1.233 also describe a general approach for identifying an appropriate scope and depth of information that applications for licenses, certifications, and approvals should provide. The content formulation should optimize the type and level of detail of information provided, based on the complexity of the design's safety case and the nexus between elements of the design and public health and safety.

In general, all applicants (especially those that do not intend to follow the LMP process) should engage the NRC staff early via pre-application activities because these interactions are an important tool for the NRC staff to plan the reviews of advanced reactor applications. While voluntary, pre-application interactions are also encouraged by the Commission as part of the NRC "Policy Statement on Advanced Reactors." These interactions with prospective applicants may be initiated once a prospective applicant has indicated sufficient commercial intent, organizational capacity, design maturity, and expectation of an application submittal to support commencement of meaningful regulatory discussions with NRC staff. The staff has issued a draft of a preapplication engagement guidance white paper (see Accession No. ML21014A267) that it is in the process of being revised. The staff intends to include the content of this white paper as Appendix C to this document.

In addition to a well-defined standard content of application methodology, the scope of information and level of detail will be informed commensurate with the type of application submitted (CP, ESP, COLA, etc.), the advanced reactor design and technology described in the application, and should also be "right-sized" based on the safety and risk significance of the structures, systems, and components associated with the facility design. The combination of the "right-sized" technical and programmatic information described above is a key component within any advanced reactor application using a risk-informed and performance based approach, but also needs to be supplemented by the safety justifications prepared by the developer, and consideration of the entirety of regulatory requirements the NRC and other agencies have established. To inform the review of the licensing basis information of a non-LWR application independent of the specific design or methodology used, the staff issued a white paper in September of 2020, titled "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors," (Ref. ML20241A017). The staff supplemented this white paper in a document dated February 2021 (Ref. ML21049A098). The September 2020, white paper as supplemented by the February 2021 document describe which regulations are generally applicable to non-LWR applications for construction permits and operating licenses under 10 CFR Part 50 and standard design certifications, combined licenses, and standard design approvals under 10 CFR Part 52.[2] The staff is in the process of revising the applicability of regulations white paper. The staff intends to include the content of this white paper as Appendix D to this document.

For example, for applicants under the 10 CFR Part 50 process, application requirements are described in 10 CFR 50.34 "Contents of applications; technical information," and for applicants under the 10 CFR Part 52 process, application requirements are described in 10 CFR 52.17 "Contents of applications; technical information," for early site permits (ESPs) and 10 CFR 52.79 "Contents of applications; technical information in final safety analysis report" for combined operating licenses (COLs).

[2] The NRC staff did not include regulations associated with early site permits, limited work authorizations, and manufacturing licenses under 10 CFR Part 52.

The Part 53 regulation is under development and as such the guidance found in this document is subject to change based on the outcome of this rulemaking. As the 10 CFR Part 53 requirements are finalized this ISG guidance will be supplemented, as necessary, to provide guidance for developing technical specifications to reflect any differences in requirements between Part 50/52 and Part 53. The goal of the 10 CFR Part 53 rulemaking effort is to develop the regulatory infrastructure to support the licensing of advanced nuclear reactors.

As a result of extensive TICAP/ARCAP public interactions with industry and external stakeholders, the proposed contents of an advanced reactor application include the following items[3]:

1. Safety Analysis Report (SAR)
2. Technical Specifications
3. Technical Requirements Manual
4. Quality Assurance (QA) Plan (design)
5. Fire Protection Program (design)
6. Probabilistic Risk Assessment
7. QA Plan (Construction and Operations)
8. Emergency Plan
9. Physical Security Plan
10. Special Nuclear Material (SNM) Control and Accountability
11. Fire Protection program (Operational)
12. Radiation Protection Program
13. Offsite Dose Calculation Manual
14. Inservice Inspection (ISI) and Inservice Testing (IST)
15. Environmental Report and Site Redress Plan
16. Financial Qualification and Insurance and Liability
17. Cyber Security Plan
18. Facility Safety Program (Under Consideration for Part 53 Applications)
19. Inspections, Tests, Analysis and Acceptance Criteria (ITAAC)

The guidance in this ISG will provide relevant information and the appropriate references for each application component identified above. Lastly, the staff notes that in order to inform the ARCAP methodology, the guidance described in this ISG leverages:

- Industry-led guidance (as endorsed),
- NRC developed guidance for advanced reactors,
- Existing guidance the NRC staff has found generally applicable, and
- Future guidance currently under development.

Subsequent revisions to this ISG will incorporate additional guidance as it is identified and developed.

[3] Submittal of certain information described on the list will be dependent on the regulatory path of an application. The list is subject to revisions including additions and should be considered preliminary.

RATIONALE

The current review guidance in NUREG-0800 "Standard Review Plan" is largely centered around large-LWRs and may not fully (or efficiently) describe a technology-inclusive, risk-informed and performance based review approach commensurate with advanced reactor technologies and the information expected to be included in an application. Based on this fact, along with the NEIMA requirements to develop a new regulatory framework, the development of a new standard content of application is warranted to ensure staff readiness to perform consistent and predictable licensing reviews. This ISG will serve as the advanced reactor application roadmap.

APPLICABILITY

This ISG applies to nuclear power reactor designers, applicants, and licensees of advanced reactors[4] (non-LWR and SMR designs) applying for permits, licenses, certifications, and approvals under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. x), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. x). It is envisioned that the review approach described in this ISG will also support the technology-inclusive, risk-informed and performance-based application content and level of detail expected in future applications submitted under the proposed Title 10 of the Code of Federal Regulations (10 CFR) Part 53, "Licensing and Regulation of Advanced Nuclear Reactors," which is currently being developed.

GUIDANCE

I. Safety Analysis Report (SAR)

An applicant for a CP should include a preliminary SAR as part of its application. Applicants for an OL or a COL should include a final SAR in their applications. Likewise, the safety analysis included in an application for an SDA, DC, or an ML should also include final information. The SAR is intended to include information that describes the facility, presents the design bases and the limits on its operation, and presents a safety analysis of the structures, systems, and components and of the facility as a whole. In general, the SAR must be sufficiently detailed to permit the staff to determine whether the plant can be built and operated without undue risk to the health and safety of the public. Prior to submission of an SAR, an applicant should have designed and analyzed the plant in sufficient detail to conclude that it can be built and operated safely. The SAR is the principal document in which the applicant provides the information needed to understand the basis upon which this conclusion has been reached and will be maintained and updated by the applicant and later as a licensee.

A twelve chapter structure for developing the SAR was discussed with industry via extensive stakeholder interactions as one acceptable approach to inform an advanced reactor application. The twelve chapter approach is largely aligned with the LMP methodology which revolves around describing the affirmative safety case at the site. Pre-application engagement between

[4] Certain elements of this RG may also be applicable to Fusion reactors, as appropriate. However, the staff notes that options for the regulatory treatment of fusion reactors are currently being considered by the NRC staff which may result in the development of fusion-specific guidance.

applicants and the NRC is highly encouraged to optimize resources and review schedule, especially for non-LMP based applications. For an advanced reactor application consistent with ARCAP/TICAP and the methodology described in RG 1.233, the 12 chapters to inform the SAR are as follows:

- Chapter 1 - General Plant Information, Site Description, and Overview of the Safety Case
- Chapter 2 - Generic Analyses
- Chapter 3 - License Basis Event Analysis
- Chapter 4 - Integrated Evaluations
- Chapter 5 - Safety Functions, Design Criteria, and SSC Categorization
- Chapter 6 - Safety Related SSC Criteria and Capabilities
- Chapter 7 - Non-Safety Related with Special Treatment (NSRST) SSC Criteria and Capabilities
- Chapter 8 - Plant Programs
- Chapter 9 - Routine Plant Radioactive Effluents, Plant Contamination, and Solid Waste
- Chapter 10 - Control of Occupational Dose
- Chapter 11 - Organization
- Chapter 12 - Initial Startup Programs.

The proposed format and content identified above is one approach to develop the contents of the SAR, but applicants have the discretion to identify alternate approaches to accommodate a variety of site conditions and plant designs. In the scenario where a particular developer uses an alternative SAR approach, particular focus should be given by the staff reviewers to any deviations and exceptions to the guidance on requested information and the organization of the information in order to ensure the required information is available to permit the staff to determine whether the plant can be built and operated without undue risk to the health and safety of the public.

<u>SAR Chapters 1-8</u>

The SAR chapters 1-8 are largely focused on describing the fundamental safety functions of the design and the affirmative safety case for each applicant consistent with the LMP approach. To inform the review of these chapters, the industry-led TICAP effort was performed and documented in NEI 21-xx. NEI 21-xx describes the scope and level of detail necessary to inform specific portions of the first 8 chapters of the SAR that are associated with the LMP-based affirmative safety case.

The NRC staff reviewed NEI 21-xx and endorsed the guidance as one acceptable approach to inform portions of the first 8 chapters of the SAR in RG 1.2XX. RG 1.2xx also describes any additional clarifications, exceptions, points of emphasis, and/or further details relevant to the specific sections discussed in NEI 21-xx. In addition, RG 1.2xx describes additional information outside the scope of LMP and NEI 21-xx that NRC staff has determined is also relevant, and would expect to be included as part of the application content related to the first 8 chapters.

Construction Permit Guidance
NEI 21-xx, Section xxx provides an acceptable method for developing portions of a construction permit application in accordance with 10 CFR Part 50 requirements. However, for advanced reactor applicants pursuing a construction permit (CP) application under 10 CFR Part 50 and using an alternative risk-informed performance-based approach, additional information not related to the LMP-based affirmative safety case should be provided. Specifically, the additional information is related to the minimum information necessary in a CP application for the staff to issue a CP under 10 CFR 50.35(a) when the applicant has not supplied all of the technical information required to complete the application (i.e., 50.34(a)) and support the issuance of a CP which approves all proposed design features (i.e., obtains finality for the design). The staff notes that the additional CP information described in RG 1.2xx provides the additional CP information necessary to supplement the first 8 Chapters of the SAR. As previously stated, the SAR chapters 1-8 are largely focused on describing the fundamental safety functions of the design and the affirmative safety case for each applicant consistent with the LMP approach.

The guidance described in Appendix E of this ISG contains guidance on one acceptable approach in scope and level of detail for applicants to provide the additional relevant CP information for advanced reactor applications. The guidance provided in Appendix E is related to Chapters 9 through 12 of the SAR, and other relevant portions of an advanced reactor application outside of the SAR.

<u>Chapter 9 - Control of Routine Plant Radioactive Effluents, Plant Contamination, and Solid Waste</u>

Overview
Nuclear power plants generate liquid, gaseous and solid waste during normal operations. Therefore, each plant must have processes to contain, store, and release these wastes in accordance with NRC regulations. In general, the information in this chapter should provide details associated with the waste management systems that ensure the requirements of 10 CFR 20, 50 and 61 are met, or propose alternative requirements consistent with the technology of the proposed advanced reactor design.

For each waste management system relied upon as part of the design, the information should include (among other things) a description related to the specific functions performed by the system, the sources of normal radioactive liquid and gaseous waste including the general quantities and composition of liquid and gaseous radioactive waste estimated to be contained in the systems, any performance monitoring of a given system, to the extent practicable, and a risk-informed approach to demonstrate compliance with the aforementioned regulations.

Staff Guidance
The guidance for the content and review, including acceptance criteria and any exceptions and clarifications are described in DANU-ISG-2021-XX, "Control of Routine Plant Radioactive Effluents, Plant Contamination and Solid Waste." (Note preliminary thoughts on potential guidance for this chapter that has not been subject to legal or management reviews are available in ADAMS at Accession No. ML20260H366).

Additional References described in the ISG

- NUREG-0800, “Standard Review Plan (SRP) for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” Sections 11.2, 11.3, and 11.4.
- Regulatory Guide (RG) 1.109 “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I”
- RG 1.111 “Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors”
- RG 4.21 “Minimization of Contamination and Radioactive Waste Generation: Life-Cycle Planning”
- NEI 07-10A “Generic FSAR Template Guidance for Process Control Program (PCP)”

<u>Chapter 10 - Control of Occupational Dose</u>

Overview
The information in this chapter should provide information on facility and equipment design, radiation sources, and operational programs that are necessary to ensure that the occupational radiation protection standards set forth in 10 CFR Part 20 are met. The information should also include any commitments made by the applicant to develop the management policy and organizational structure necessary to ensure occupational radiation exposures are as low as (is) reasonably achievable (ALARA).

Staff Guidance
The guidance for the content and review, including acceptance criteria and any exceptions and clarifications are described in DANU-ISG-2021-XX, “Control of Occupational Dose.” (Note preliminary thoughts on potential guidance for this chapter that has not been subject to legal or management reviews are available in ADAMS at Accession No. ML20260H366).

Additional References described in the ISG
- RG 8.8 “Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable”
- RG 8.10 “Operating Philosophy for Maintaining Occupational Radiation Exposures as Low as Is Reasonably Achievable”
- ANSI/ANS 18.1-1999 “Radioactive Source Term For Normal Operation Of Light Water Reactors”
- NEI 07-08A “Generic FSAR Template Guidance for Ensuring that Occupational Radiation Exposures are as Low as is Reasonably Achievable (ALARA)”

<u>Chapter 11 – Organization</u>

Overview
The information in this chapter should provide descriptions of the organizational structure and key management positions within the design, construction and operating organizations that are responsible for facility design, design review, design approval, construction management, testing, and operation of the plant.

Staff Guidance

The guidance for the content and review, including acceptance criteria and any exceptions and clarifications are described in DANU-ISG-2021-XX, "Organization." (Note preliminary thoughts on potential guidance for this chapter that has not been subject to legal or management reviews are available in ADAMS at Accession No. ML21049A277).

Additional References described in the ISG

Chapter 12 - Initial Startup Programs

Overview
The information in this chapter should provide a description of the Initial Startup Program (ISP) in the application. The ISP consists of preoperational testing (i.e. tests conducted following construction and construction related testing, but prior to initial fuel load) and initial startup testing (i.e. tests conducted during and after initial fuel load, up to and including initial power ascension). The primary objective of the ISP is to is to demonstrate, to the extent possible, that the safety-related (SR), safety-significant (SS) and radiation monitoring SSCs operate in accordance with the design and as assumed in the safety analysis.

Staff Guidance
The guidance for the content and review, including acceptance criteria and any exceptions and clarifications are described in DANU-ISG-2021-XX, "Initial Startup Program." (Note preliminary thoughts on potential guidance for this chapter that has not been subject to legal or management reviews are available in ADAMS at Accession No. ML21049A277).

Additional References described in the ISG
- NUREG-0800 (SRP) Sec. 14.2

II. Technical Specifications

Overview
In general, Technical Specifications (TS) are part of an NRC license authorizing the operation of a nuclear production or utilization facility. A technical specification establishes requirements for items such as safety limits, limiting safety system settings, limiting control settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

Section 182a of the Atomic Energy Act of 1954, as amended, requires applicants for nuclear power plant operating licenses to state the following:

> *[S]uch technical specifications, including information of the amount, kind, and source of special nuclear material required, the place of the use, the specific characteristics of the facility, and such other information as the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization...of special nuclear material will be in accord with the common defense and security and will provide adequate protection to the health and safety of the public. Such technical specifications shall be a part of any license issued.*

In Title 10 of the Code of Federal Regulations (10 CFR) Section 50.36, "Technical Specifications," the Commission established its regulatory requirements related to the content of TS. For an advanced reactor application, the NRC staff may need to review and assess proposed TS by applicants using a risk-informed evaluation process

Staff Guidance
The guidance for the content and review, including acceptance criteria and any exceptions and clarifications are described in DANU-ISG-2021-XX, “Risk-Informed Technical Specifications.” (Note preliminary thoughts on potential guidance for this chapter that has not been subject to legal or management reviews are available in ADAMS at Accession No. ML21133A490).

Additional References described in the ISG

III. Technical Requirements Manual

Staff is still considering whether guidance in this area is necessary. Further discussion is warranted.

IV. Quality Assurance Plan (design)

Staff is still considering whether guidance in this area is necessary. Further discussion is warranted.

V. Fire Protection Program (design)

The LMP process includes limited fire protection information. Need to assess what additional information (if any) is necessary. Staff is considering whether guidance in this area is necessary. Further discussion is warranted.

Additional References
- RG 1.189 “Fire Protection for Nuclear Power Plants”

VI. Probabilistic Risk Assessment

This section is currently a placeholder. The NRC staff is evaluating the need for, and contents of this section in addition to what has already been discussed in SAR Chapter 2 and relevant results of the PRA in Chapters 3 through 7. The NRC staff is also considering other ongoing activities (see Additional References below) as part of the evaluation.

Additional References
- Draft NRC white paper “Demonstrating the Acceptability of Probabilistic Risk Assessment Results Used to Support Advanced Non-Light Water Reactor Plant Licensing” (ML21015A434)
- ASME/ANS RA-S-1.4-2021 “Probabilistic Risk Assessment Standard for Advanced Non-Light Water Reactor Nuclear Power Plants”
- NEI 20-09, Revision 1, “Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard” (ML20302A115)
- Action Plan, “Review and Endorsement of ASME/ANS Advanced Non-LWR PRA Standard ASME/ANS RA-S-1.4” (ML20104C132)

VII. Quality Assurance (QA) Plan (Construction and Operations)

Staff is still considering whether guidance in this area is necessary. Further discussion is warranted.

Additional References
- RG 1.28 “Quality Assurance Program Criteria (Design and Construction)”
- RG 1.30 “Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Safety Guide 30)”
- RG 1.33 “Quality Assurance Program Requirements (Operation)”
- RG 1.164 “Dedication of Commercial-Grade Items for Use in Nuclear Power Plants”
- QA Plan for sodium-cooled FAST Metallic Fuel Data Qualification

VIII. Emergency Preparedness Plan

Overview
The ongoing “Emergency Preparedness Requirements for Small Modular Reactors and Other New Technologies” rulemaking would amend the NRC’s regulations to add new emergency preparedness requirements for small modular reactors and other new technologies such as non-light-water reactors and non-power production or utilization facilities. The rule would adopt a scalable plume exposure pathway emergency planning zone approach that is performance-based, consequence-oriented, and technology-inclusive. This rulemaking would affect applicants for new NRC licenses and reduce regulatory burden related to the exemption process.

Staff Guidance
The guidance for the content and review, including acceptance criteria and any exceptions and clarifications are described in draft RG DG-1357, “Emergency Response Planning and Preparedness for Nuclear Power Reactors.” Upon completion of the final RG, this section will be updated.

Additional References
- NUREG-0396 “Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants”
- NUREG-0654 “Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants (FEMA-REP-1)”
- RG 1.101 “Emergency Planning and Preparedness for Nuclear Power Reactors”
- SRM-SECY-16-0069
- SECY-18-0103 related to EP for SMRs and other technologies

IX. Physical Security Plan

Overview
Physical security rulemaking expected to develop guidance in this area.

Additional References

- SECY-18-0076 “Options and Recommendation for Physical Security For Advanced Reactors”

X. Special Nuclear Material (SNM) Control and Accountability

Staff is still considering whether guidance in this area is necessary. Further discussion is warranted.

Additional References for discussion

- Check NUREG-2159 for applicability
- Check Advanced Reactors integrated schedule list of reports

XI. Fire Protection program (Operational)

Need to assess what additional information (if any) is necessary. *Staff is considering whether guidance in this area is necessary. Further discussion is warranted.* Consideration for fire protection guidance for operational programs include:

- Part 53 is considering developing a fire protection operational program regulation

- For Part 50 and 52 applications for non-LWRs the applicability of regulations white paper described above notes that 10 CFR 50.48(a), “fire protection plan,” and GDC 3/ARDC 3 apply to non-LWRs. This paper also notes that 10 CFR 50.48(b), 10 CFR Part 50 Appendix R, and 50.48(c), do not apply to non-LWRs

- For non-LWRs that have coolants that could include a fire hazard, does additional fire protection operational program guidance need to be developed.

Additional References

- RG 1.189 “Fire Protection for Nuclear Power Plants”

XII. Radiation Protection program

Staff is still considering whether guidance in this area is necessary. Further discussion is warranted.

XIII. Offsite Dose Calculation Manual

Staff is still considering whether guidance in this area is necessary. Further discussion is warranted.

XIV. ISI/IST

TICAP outcomes expected to heavily influence ISI/IST. In addition, ASME Section XI Section 2 guidance identified as needing to be developed. The link to TICAP is through special treatment requirements. TICAP does not intend to develop guidance on documenting ISI/IST programs. *Staff is developing guidance for this issue.*

Additional References

- DG 1383
- RG 1.178 "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping"
- Guidance is forthcoming (INL ISG)

XV. Environmental Report and Site Redress Plan

Staff is still considering whether additional guidance in this area is necessary. Further discussion is warranted.

Additional References

- RG 4.2 "Preparation of Environmental Reports for Nuclear Power Stations"
- NUREG-1555 "Standard Review Plans for Environmental Reviews for Nuclear Power Plants: Environmental Standard Review Plan (with Supplement 1 for Operating Reactor License Renewal)"
- COL/ESP-ISG-026
- COL/ESP-ISG-027
- Environmental ISG for Micro Reactors
- Draft GEIS for Adv. Rxs

XVI. Financial Qualification and Insurance and Liability

Additional References

- Price-Anderson Report under development to address issues. Expected to be completed by end of 2021. Additional details in the Advanced Reactor Integrated Schedule.

XVII. Cyber Security Plan

Staff is still considering whether additional guidance in this area is necessary. Further discussion is warranted.

Additional References

- RG 5.71 "Cyber Security Programs for Nuclear Facilities"

XVIII. Facility Safety Program

This section is specific to 10 CFR 53 and does not apply to 10 CFR 50 or 10 CFR 52 applicants. Need to develop guidance in this area is dependent on the outcome of Commission direction on proposed rulemaking.

XIX. Inspections, Tests, Analysis, and Acceptance Criteria

ISG under development will address ITAAC. Further discussion is warranted.

IMPLEMENTATION

The staff will use the information discussed in this ISG to determine the following:

[Identify how the information will facilitate staff review of license amendments, license renewal applications, etc.]

BACKFITTING AND ISSUE FINALITY DISCUSSION

[OGC provides this discussion, but the staff can propose text for OGC consideration].

Example: The NRC staff issuance of this ISG is not considered backfitting as defined in 10 CFR 50.109(a)(1), nor is it deemed to be in conflict with any of the issue finality provisions in 10 CFR Part 52.

CONGRESSIONAL REVIEW ACT

[OGC provides this discussion to support issuance of the final ISG. However, the staff can propose text for OGC consideration].

Example: This ISG is a rule as defined in the Congressional Review Act (5 U.S.C. §§ 801-808). However, the Office of Management and Budget has not found it to be a major rule as defined in the Congressional Review Act.

FINAL RESOLUTION

By [insert date], this information will be transitioned into [identify the appropriate regulatory process (Standard Review Plan (SRP), Regulatory Guide (RG))]. Following the transition of this guidance to the [SRP, RG], this ISG will be closed.

APPENDIXES

A. Resolution of Public Comments
B. References
C. Pre-Application Engagement Guidance
D. Analysis of Applicability of NRC Regulations for Non-Light Water Reactors
E. Construction Permit Guidance

APPENDIX A

Resolution of Public Comments

A notice of opportunity for public comment on this Interim Staff Guidance (ISG) was published in the *Federal Register* (*insert FR Citation #)* on [date] for a 30-60 day comment period. [Insert number of commenters] provided comments which were considered before issuance of this ISG in final form.

Comments on this ISG are available electronically at the NRC's electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can gain entry into ADAMS, which provides text and image files of NRC's public documents. Comments were received from the following individuals or groups:

Letter No.	ADAMS No.	Commenter Affiliation	Commenter Name	Abbreviation
1				
2				
3				
4				
5				

The comments and the staff responses are provided below.

Comment 1: [Each comment summary must clearly identify the entity that submitted the comment and the comment itself].

NRC Response: Comment responses should begin with a direct statement of the NRC staff's position on a comment, e.g., "the NRC staff agrees with the comment" or the "NRC staff disagrees with the comment".

- If the NRC staff agrees, explain why and provide a clear statement as to how the relevant language was revised or supplemented to address the comment. Include the following language at the end of the comment response: "The final ISG was changed by <*describe the change; if necessary, by quoting the newly revised language*>."
- If the NRC disagrees with a comment and no change was made to the generic communication, then explain why and provide the following language at the end of the comment response: "No change was made to the final ISG as a result of this comment."

APPENDIX B

References

APPENDIX C

Pre-Application Engagement Guidance

APPENDIX D

Analysis of Applicability of NRC Regulations for Non-Light Water Reactors

Appendix E
Construction Permit Guidance

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC, or Commission) staff is providing this guidance to facilitate discussion of the safety review of non-light water reactor (non-LWR) construction permit (CP) applications for power reactors. Note that this Construction Permit Guidance Section is a follow-on to a white paper on the topic. The draft white paper “Safety Review of Power Reactor Construction Permit Applications” be found in ADAMS Accession No. ML21043A339. The white paper included CP guidance for both LWRs and non-LWRs. The NRC staff has determined that going forward it is best to split the CP guidance into separate guidance for LWRs and non-LWRs. However, the staff recognizes that there is a portion of the guidance that is applicable to both types of designs. Portions of the guidance that is applicable to both LWRs and non-LWRs is shown in italics below. The information in italics will be updated as the LWR and non-LWRs guidance is further refined.

BACKGROUND

The NRC anticipates the submission of power reactor CP applications within the next few years. The review of these applications falls within the two-step licensing process under 10 CFR Part 50 and involves the issuance of a CP before an operating license (OL). The NRC last reviewed a power reactor CP in the 1970s. Most recently, the NRC issued combined construction and operating licenses (combined licenses) for power reactors through the one-step licensing process under 10 CFR Part 52 utilizing guidance in the Standard Review Plan (SRP, NUREG-0800) (Ref. 8) and Regulatory Guide (RG) 1.206 (Ref. 17, 18).

The licensing process under 10 CFR Part 50 allows an applicant to begin construction with preliminary design information as compared with the final design required for a combined license (COL) under 10 CFR Part 52. Although the two-step licensing process provides flexibility and a more limited safety review prior to construction, there is less finality on the design before the applicant commits to construction of the facility.

The SRP contains the staff review guidance for LWR applications submitted under 10 CFR Part 50 or 10 CFR Part 52. In addition, some insights on the level of detail that is required for the preliminary safety analysis report (PSAR) in support of the CP application may be obtained from RG 1.70, Revision 3, 1978, (Ref. 13) but these insights may be limited to the degree that the guidance does not account for subsequent requirements and NRC technical positions, or advances in technical knowledge. RG 1.206 provides guidance for COL applications and includes insights on the level of detail needed for final design information if the CP applicant chooses to provide such information.

The NRC is developing guidance for the safety review of non-LWR designs. The Advanced Reactor Content of Applications Project (ARCAP) document will reference existing guidance that may be applicable to non-LWR designs and recently developed non-LWR guidance for specific areas of review. The ARCAP is broader and encompasses the industry-led Technology-Inclusive Content of Application Project (TICAP). These projects build on the outcome of the Licensing Modernization Project (LMP), which provides guidance that focuses *on identifying licensing basis events; categorizing and establishing performance criteria for structures, systems, and components; and evaluating defense in depth for advanced reactor designs.*

ARCAP guidance is being developed independently of the SRP for light water reactors. Because ARCAP guidance is envisioned to use an application structure different than the SRP, Appendix C, "Advanced Reactor Construction Permit Guidance," has been developed for applications that choose to follow this approach.

The NRC recently issued CPs for two non-power production and utilization facilities, SHINE Medical Isotopes (Ref. 9) and Northwest Medical Isotopes (Ref. 10). Some of the lessons learned from these reviews are applicable to the review of power reactor CP applications and are summarized below.

RATIONALE

During the June 12, 2020, public meeting on the Advanced Reactor Content of Application Project for non-LWR designs, the Nuclear Energy Institute (NEI) and U. S. Nuclear Industry Council (USNIC) requested guidance for CP applicants within the next 1-2 years.

In a subsequent public meeting on July 31, 2020, the staff presented options to address industry's request to support the timeline of potential applications and received feedback that the interim staff guidance (ISG) option appears to address industry's needs for near-term CP guidance.

This draft white paper focuses on the safety review of power reactor CP applications and may be further developed into an ISG applicable to any LWR design, including designs similar to those recently reviewed under 10 CFR Part 52, and may refer to the applicable guidance for the review of non-LWR designs. It has been approximately 40 years since the staff reviewed a CP application for a power reactor. Although the LWR CP application guidance in RG 1.70 dates from the 1970s and the more recent LWR application guidance in RG 1.206 was developed for a COL application, these documents provide some insights on the level of detail to support an LWR CP application review as discussed above. For a non-LWR CP application, the ARCAP guidance provides information on the level of detail to meet the applicable requirements for a CP.

This draft white paper also includes a discussion of how the staff's safety review would address LWR applications that reference an approved design or other NRC approvals, specific CP safety review areas needing clarity, and applicability of ARCAP guidance.

GUIDANCE

Requirements for a Power Reactor Construction Permit Application

A number of regulations apply to a power reactor CP application, including:

- *10 CFR 50.30, "Filing of application; oath or affirmation"*

- *10 CFR 50.33, “Contents of applications; general information”*[5]
- *10 CFR 50.34, “Contents of applications; technical information,” particularly paragraph (a), “Preliminary safety analysis report,”*
- *10 CFR 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents – nuclear power reactors”*
- *10 CFR 50.35, “Issuance of construction permits”*
- *10 CFR 50.40, “Common standards”*
- *10 CFR 50.55, “Conditions of construction permits, early site permits, combined licenses, and manufacturing licenses”*
- *10 CFR 50.55a, “Codes and standards”*
- *10 CFR Part 20, “Standards of Protection Against Radiation”*
- *10 CFR Part 100, “Reactor Site Criteria”*

The regulations in 10 CFR 50.34(a) specify the minimum technical information in the preliminary safety analysis report (PSAR) accompanying a CP application, including a description and safety assessment of the site on which the facility is to be located. The site safety assessment is expected to include an analysis and evaluation of the major structures, systems and components (SSCs) of the facility that bear significantly on the acceptability of the site under the site evaluation factors identified in 10 CFR Part 100.

The regulations in 10 CFR 50.35, “Issuance of construction permits,” provide for the issuance of a CP in cases where the application does not provide sufficient information for the staff to approve all proposed design features and when certain criteria are met. In its early practices, the predecessor to the NRC, the Atomic Energy Commission (AEC), had issued a “provisional” CP when the applicant had not submitted all the technical information to complete the application and to approve all proposed design features. However, almost all issued “provisional” CPs were never converted to a “final” CP but instead merged into an operating license. Therefore, the AEC proposed to codify the Commission’s practice for issuing a CP (34 FR 6540, April 16, 1969). The final amendment to the regulations in 10 CFR 50.35 eliminated the term “provisional” construction permit but retained the “provisional” criteria for issuing a CP (35 FR 5317, March 31, 1970). By issuing a CP, the Commission authorizes the construction of the facility described in the application, including the principal architectural and engineering criteria and identification of major features or components for the protection of the health and safety of the public.

The current regulations for issuing a CP in 10 CFR 50.35(a) have not been modified since 1970:

> *(a) When an applicant has not supplied initially all of the technical information required to complete the application and support the issuance of a construction permit which approves all proposed design features, the Commission may issue a construction permit if the Commission finds that (1) the applicant has described the proposed design of the facility, including, but not limited to, the principal architectural and engineering criteria for the design, and has identified the major features or components incorporated therein for the protection of the health and safety of the public; (2) such further technical or design*

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[5] Although referenced herein, guidance on compliance with the applicable requirements in 10 CFR 50.30 and 50.33 is outside the scope of this document.

information as may be required to complete the safety analysis, and which can reasonably be left for later consideration, will be supplied in the final safety analysis report; (3) safety features or components, if any, which require research and development have been described by the applicant and the applicant has identified, and there will be conducted, a research and development program reasonably designed to resolve any safety questions associated with such features or components; and that (4) on the basis of the foregoing, there is reasonable assurance that, (i) such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and (ii) taking into consideration the site criteria contained in part 100 of this chapter, the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

In cases where a novel design has not sufficiently progressed and certain information is not available at the submission of the CP application, the PSAR should provide the criteria and bases used to develop the required information, the concepts and alternatives under consideration, and the schedule for completion of the design and submission of the missing information. In general, the PSAR should describe the preliminary design of the facility in sufficient detail to enable the staff to evaluate whether the facility can be constructed and operated without undue risk to the health and safety of the public.

The criteria in 10 CFR 50.35(a) focus on the safety aspects of the design, including the principal architectural and engineering criteria and the safety design features, as well as siting information to support construction of the facility. Given the advances in technology since the most recent amendment of the regulation, it may be easier for an applicant to provide more complete technical information in its CP application and thereby reduce the regulatory review in the subsequent licensing phase. As noted in 10 CFR 50.35(a), the findings above will be modified, if specifically requested by the applicant, for a complete CP application that includes all technical information, including the final design of the facility.

Under 10 CFR 50.35(b), a CP applicant may also request approval of any design features or specifications in its CP application. This request for approval would need more than preliminary information to support the staff's review to approve such design features or specifications. In such a case it would be expected that the level of design information available to support the approval of a proposed design feature in the application would be the same level of design information available for a 10 CFR Part 52 COL application. Guidance for the expected level of design information that is available to support a COL application can be found in RG 1.206. It should be noted that any approval, if granted, would apply only to the extent that the item has been fully addressed or treated in the application and would not extend beyond items or details not fully covered in the application. The regulation at 10 CFR 50.35(b) clarifies that a CP authorizes the applicant to proceed with construction but is not an approval of the safety of any design features or specifications unless the applicant requests for such approval and the approval is incorporated into the permit.

As described in 10 CFR 50.35(c), a license authorizing operation of the facility will not be issued until (1) the applicant submits, as part of an OL application, its final safety analysis report (FSAR) and (2) the Commission finds that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility. The FSAR *submitted with the OL should describe in detail the final design of the facility as constructed, identify the changes from the criteria, design, and bases in the PSAR, and discuss the bases and safety significance of the changes from the PSAR. Prior to the issuance of an operating license, the staff will review the applicant's final design in the FSAR to determine whether all the Commission's safety requirements have been met. Based on this determination, the Commission would issue an OL and the applicant may then operate the facility in accordance with the terms of the OL and the Commission's regulations under the continued oversight by the NRC staff.*

<u>Lessons Learned from Recently Issued CPs</u>

Recently, the NRC issued permits for the construction of medical radioisotope facilities as non-power production and utilization facilities (NPUFs) licensed under 10 CFR Part 50. The Commission issued CPs to SHINE Medical Technologies, LLC in February 2016, and Northwest Medical Isotopes, LLC in May 2018. Lessons learned from the review of these NPUF CP applications include the following:

- *Pre-application engagement is key to providing near-term guidance to the applicant.*
- *Early interactions supported common understanding of what information is needed in the PSAR and what information could be reasonably left for the FSAR accompanying the OL application, e.g., operational program descriptions.*
- *If the PSAR includes preliminary or limited descriptions of the facility's programs, structures, systems, or components, the staff may accept and approve the application with regulatory commitments from the applicant to provide complete information in its OL application.*
- *The staff's construction permit safety review is focused on ensuring appropriate use of analysis methodologies to meet the requirements in the regulations.*

In safety evaluations related to the CPs issued, the NRC staff noted applicant regulatory commitments regarding the resolution of items that were not necessary for the issuance of a construction permit, but that the applicant should address in the FSAR submitted with an operating license application. The CPs included conditions to ensure that the permit holder informed the NRC of safety significant areas of construction prior to the submission of an OL application. CP conditions of a confirmatory nature focused on additional information needed to address certain matters related to the safety of a final design and required the applicant to submit, prior to the completion of construction, periodic reports on such information to the NRC.

The NPUF lessons learned noted above may be applied for an effective and efficient safety review of the PSAR to determine whether the application meets the 10 CFR 50.35 requirements for issuing a CP. However, in drawing lessons from recent NPUF reviews, consideration should be given to the different technologies involved and the much more limited set of safety requirements that apply to an NPUF as opposed to a power reactor.

Consistent with past practice and experience, including the recent NPUF reviews discussed above, pre-application activities have proven effective in gaining early understanding of the applicant's plans and its proposed facility design, supporting early resolution of unique design aspects of the facility, and preparing resources for the review of the application. Also, a recent *staff draft white paper (Ref. 5) on preapplication engagement to optimize application reviews provides information to advanced reactor developers on the benefits of robust preapplication engagement in order to optimize application reviews. Although directed to the advanced reactor community, the draft white paper describes a set of pre-application activities that may be applicable to LWR license applicants and, if fully executed, will enable the staff to offer more predictable and shorter schedules and other benefits during the review of a reactor license application.*

Special Topics

The previous section provides guidance on the overall approach for the safety review of a CP application recognizing that if an application does not provide the information to support the issuance of a construction permit that approves all proposed design features, it may still meet the criteria in 10 CFR 50.35(a) for the Commission to issue a CP.

This section provides additional guidance on potential CP application submissions and the effect of ongoing regulatory activities on the review of future CP applications.

Concurrent Applications

A CP application may be accompanied by an application for a limited work authorization (LWA). For the LWA review, the staff should refer to the guidance in COL/ESP-ISG-4 (Ref. 7) related to the definition of construction and limited work authorization.

Questions have been raised regarding the possibility of submitting the OL application before the CP is issued. The staff is still considering the legal and policy implications of this possibility. For an OL application submitted before the construction permit is issued, a process would need to be developed to address the CP mandatory hearing (if not completed before the OL application is submitted) and the logistics associated with the OL hearing opportunity.

The staff notes that there are inherent complications associated with a concurrent CP and OL review. For example, as a result of the OL review, a need to reclassify SSCs (i.e., from non-safety-related to safety-related) could arise based on updated design information that was not available at the time of the CP. In such a case, extensive rework of both the CP and OL applications could be needed to address this reclassification.

CP Application Incorporating Prior NRC Approvals

A CP application may incorporate prior NRC approvals by reference, including a standard design approval (SDA), a certified design (DC), or an early site permit (ESP). Each of these approvals is supported by a staff safety evaluation concluding that the applicant has met the specific regulatory requirements for approval and may be subject to conditions and additional requirements and restrictions. These prior NRC approvals have finality when referenced in a CP application as defined by the issue finality provisions for the particular Part 52 approval.

If the staff determines that the CP application demonstrates the applicability of the prior NRC approval including compliance with any associated conditions and additional requirements and restrictions, the staff's CP review regarding the referenced material would generally be limited to *an evaluation of (1) how the referenced approval conditions and additional requirements and restrictions are addressed in the CP application, and (2) any deviations from the referenced material that are subject to prior NRC review. Portions of the application not receiving prior NRC approval will be the focus of the NRC staff's CP review.*

For a CP application referencing an ESP, the staff's review and evaluation may be more extensive in that the staff would conduct a safety review and evaluation of the proposed design of the facility, any requested variances from the ESP, the satisfaction of any relevant permit conditions, and the updating of emergency preparedness information in accordance with 10 CFR 52.39(b). As provided by 10 CFR 52.24(b), any ESP terms or conditions that cannot be met by CP issuance must be set forth as terms or conditions of the CP.

For a CP application referencing an SDA or DC, the staff's review and evaluation may be focused on the suitability of the selected site for the referenced design, the satisfaction of any additional requirements or restrictions for the approved design, and any design matters outside the scope of the referenced design. Under 10 CFR Part 52, a DC must be based on essentially complete design, while an SDA may approve only major features of the design; this difference may affect the level of design information that might be needed in the CP application. Also, Section IV.B in all issued design certification rules provides that "[t]he Commission reserves the right to determine in what manner this appendix may be referenced by an applicant for a construction permit or operating license under 10 CFR part 50."

For a CP application referencing an ESP and an SDA or DC, the staff's review and evaluation would generally be focused on whether the referenced design fits within the characteristics of the approved site; whether the other applicable conditions, requirements, and restrictions in the referenced approvals are satisfied; whether deviations from the referenced approvals that require prior NRC approval comply with NRC regulations; and whether requirements for matters outside the scope of the referenced approvals are met.

Ongoing Regulatory Activities

The NRC is currently pursuing the alignment of requirements in 10 CFR Parts 50 and 52 through rulemaking. The rulemaking is in its initial phases and may include additional licensing requirements for applications submitted under 10 CFR Part 50 (e.g., risk information). Until the final rule is issued, a CP application will be reviewed and evaluated in accordance with the existing regulations. The staff should continue to monitor the progress of the 10 CFR Parts 50 and 52 rulemaking since a CP applicant must comply with the applicable regulations that are in effect at the time the NRC issues the construction permit. A CP applicant may choose to provide risk information in its application and the staff should consider this information to enhance its review focus on the proposed safety design features of the facility.

The NRC is working on the advanced reactor content of application project (ARCAP) to develop technology-inclusive, risk-informed, and performance-based application guidance. The ARCAP guidance is intended for use by an advanced reactor applicant for a combined license, construction permit, operating license, design certification, standard design approval, or manufacturing license. Many of the topics covered in the ARCAP guidance may also be applicable to LWR designs, including updated siting guidance. The staff should consider the *updated guidance in the ARCAP, when finalized, for applicability to a CP application review as described in Appendix C of this document.*

Receipt, Possession, and Use of Source, Byproduct and Special Nuclear Material

This document does not provide guidance on the licensing requirements for byproduct, source, or special nuclear material under 10 CFR Parts 30, 40, and 70. The CP applicant may address the applicable materials licensing requirements with its CP application (in accordance with 10 CFR 50.31) or separately from the CP application.

Detailed Advanced Reactor Construction Permit Guidance

This portion of the construction permit (CP) content guidance is intended for CP applications involving advanced non-light water reactors (LWRs). The guidance is based on an application using a risk-informed performance-based approach, such as the advanced reactor content of application project (ARCAP) whose purpose is to develop technology-inclusive, risk-informed and performance-based application guidance. The ARCAP, documented in *ISG-XXX, "Advanced Reactor Content of Application Interim Staff Guidance,"* is broad and encompasses the industry-led technology-inclusive content of application project (TICAP). This CP guidance references applicable guidance developed through the ARCAP/TICAP activities as well as guidance derived from separate ongoing regulatory activities (e.g., security and emergency planning rulemaking), as necessary.

The TICAP guidance that is being developed in parallel with the guidance found in this document is based on the Licensing Modernization Project (LMP) as endorsed by Regulatory Guide (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." Several vendors have indicated that they plan to implement the LMP to develop the licensing basis for their applications. As such, processes from the LMP and initial guidance referencing TICAP and ARCAP draft documents are referenced throughout this document.

The ARCAP guidance is currently under development and is intended to be used in conjunction with the guidance in this document for the review of a non-LWR CP application. Because ARCAP/TICAP is in its early stages this document italicizes NRC guidance and industry standards that are under development that are not yet formally endorsed. These italics will be removed in future revisions to the document as the ARCAP/TICAP guidance and other NRC guidance and Industry standards to reflect the appropriate endorsed guidance.

However, applicants are not required to utilize the TICAP/LMP approach and may instead use another methodology (e.g., traditional deterministic approach, maximum hypothetical accident[6]) to analyze non-LWR performance and develop a licensing basis. The TICAP/LMP process forms the basis for this guidance although in some areas the guidance provides additional considerations for acceptably addressing a specific topic when a TICAP/LMP approach is not

[6] In this context, "maximum hypothetical accident" refers to a conservatively assessed, deterministic accident with consequences that bound the full spectrum of accident conditions for the plant and is not necessarily a credible event.

used. As noted above applicants are encouraged to use the preapplication process to optimize reviews, which is especially important if an applicant intends to use a process other than the LMP to develop their licensing basis. Regardless, the review guidance in this document is limited in scope. NRC staff should continue to consult other established guidance documents, as applicable, to complete reviews of non-LWR applications.

This guidance addresses the minimum information necessary in a CP application for the staff to issue a CP under 10 CFR 50.35(a) when the applicant has not supplied all of the technical information required to complete the application (i.e., 50.34(a)) and support the issuance of a CP which approves all proposed design features (i.e., obtains finality for the design). When making its safety finding regarding the issuance of a CP under 50.35(a), the staff should make the determination that the application:

(1) Describes the proposed design of the facility, including, but not limited to,
  a. the principal architectural and engineering criteria for the design, and
  b. the major features or components incorporated therein for the protection of the health and safety of the public.

(2) Describes safety features or components, if any, which require research and development program necessary to resolve any safety questions associated with such features or components.

(3) Provides commitments that such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and

(4) Describes the site criteria contained in 10 CFR Part 100 and based on that criteria concludes that the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

Where an applicant desires design finality regarding a specific topic, the NRC staff should review that the application has provided sufficient information about the topic at a level of detail that is expected at the operating license (OL) stage. Refer to *the draft TICAP ISG and draft ARCAP ISG*.

Specific Topic Guidance

Chapters 1-8

NEI 21-xx, Section xxx provides an acceptable method for developing portions of a construction permit application in accordance with 10 CFR Part 50 requirements. However, for advanced reactor applicants pursuing a construction permit (CP) application under 10 CFR Part 50 and using an alternative risk-informed performance-based approach, additional information not related to the LMP-based affirmative safety case should be provided. Specifically, the additional information is related to the minimum information necessary in a CP application for the staff to issue a CP under 10 CFR 50.35(a) when the applicant has not supplied all of the technical information required to complete the application (i.e., 50.34(a)) and support the issuance of a CP which approves all proposed design features (i.e., obtains finality for the design). The staff notes that the additional CP information described in RG 1.2xx provides the additional CP information necessary to supplement the first 8 Chapters of the SAR. As previously stated, the SAR chapters 1-8 are largely focused on describing the fundamental safety functions of the design and the affirmative safety case for each applicant consistent with the LMP approach.

9. Control of Routine Plant Radioactive Effluents, Plant Contamination and Solid Waste
   For guidance regarding specific information content refer to *draft ARCAP ISG, "Control of Routine Plant Radioactive Effluents, Plant Contamination and Solid Waste."* (Note preliminary thoughts on potential guidance for this chapter that has not been subject to legal or management reviews are available in ADAMS at Accession No. ML20260H366).

10. Control of Occupational Dose
    For guidance regarding specific information content refer to *draft ARCAP ISG, "Control of Occupational Dose."* (Note preliminary thoughts on potential guidance for this chapter that has not been subject to legal or management reviews are available in ADAMS at Accession No. ML20260H366).

11. Organization

    For guidance regarding specific information content refer to *draft ARCAP ISG, "Organization."* (Note preliminary thoughts on potential guidance for this chapter that has not been subject to legal or management reviews are available in ADAMS at Accession No. ML21049A277).

12. Initial Startup Program
    For guidance regarding specific information content refer to *draft ARCAP ISG, "Initial Startup Program." (Note preliminary thoughts on potential guidance for this chapter that has not been subject to legal or management reviews are available in ADAMS at Accession No. ML21049A277).*

13. Quality Assurance
    The staff should review the applicant's quality assurance program description (QAPD) applied to activities for design, fabrication, construction, and testing of the safety-related and safety-significant SSCs of a facility or facilities that may be constructed on the site. The staff should approve the QADP prior to the start of included activities.

    The staff's review should ensure that the applicant (and its principal contractors such as the reactor vendor, Architect Engineer, constructor and construction manager) has established a QA program for the design and construction phases in accordance with Appendix B to 10 CFR Part 50, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." The QA program should also address the collection of site information. The applicant's QA program (including its principal contractors) must describe in the CP application how each criterion of Appendix B will be met or propose an alternate or limited set of criteria with appropriate justifications. The staff should expect to review applicant submitted exemption requests where alternate requirements are being proposed to the Appendix B regulations.

    The staff should refer to the guidance in RG 1.28, "Quality Assurance Program Criteria (Design and Construction)," as an acceptable approach to establishing and implementing a QA program for the design and construction of nuclear power plants. This RG endorses, with certain exceptions and clarifications, the Part I and Part II requirements included in the NQA-1b-2011 Addenda to ASME NQA-1-2008, NQA-1-2012, and NQA-1-2015, "Quality Assurance Requirements for Nuclear Facility Applications," for the implementation of a QA program during the design and construction phases of nuclear power plants that provides an adequate basis for complying with the requirements of Appendix B to 10 CFR Part 50.

NRC SECY-03-0117, "Approaches for Adopting More Widely Accepted International Quality Standards," documents the staff's effort to review international quality assurance standards against the existing 10 CFR Part 50 Appendix B framework and assess approaches for adopting international quality standards for safety-related components in nuclear power plants into the existing regulatory framework. The staff should refer to this document when reviewing an application that uses international QA standards to meet 10 CFR Part 50 Appendix B requirements.

14. Security
The staff should review the application to verify that it contains the following information:
    a. Information demonstrating that site characteristics are such that adequate security plans and measures can be developed consistent with the guidance in *draft ARCAP ISG, section 2.1, "Site Characteristics and Site Parameters (Overview),"* (note that no Physical Security Plan, Security Training and Qualifications Plan, or Safeguards Contingency Plan information is required at the CP stage).
    b. Information Security Plan – the application should include a plan for the protection of safeguards information (SGI). This plan should be reviewed and approved by NRC during the preapplication period to enable the NRC staff to provide the applicant with SGI documents, as necessary, for the applicant to consider safeguards and security in the design of the facility, development of the physical security program to meet the requirements of 10 CFR Part 73, "Physical Protection of Plants and Materials," and address safety concerns associated with 10 CFR 50.150, "Aircraft impact assessment," in their application.

15. Emergency Planning
The NRC staff should review the application to verify that it contains the following information:
    a. Describe any physical characteristics of the proposed site, such as egress limitations from the area surrounding the site, that could pose a significant impediment to the development of emergency plans (EPs) (note that no EP is required at the CP stage). If physical characteristics are identified that could pose a significant impediment to the development of EPs, the application should identify measures that would, when implemented, mitigate or eliminate the significant impediment.

    b. Describe the major features of the EP which are aspects of the plan necessary to:

i. Address in whole or part either one or more of the 16 standards in 10 CFR 50.47(b) or the proposed requirements of 10 CFR 50.160(b)[7], as applicable; or
ii. Describe the emergency planning zones as required in 10 CFR 50.33(g).

Refer to *draft Regulatory Guide (DG), DG–1350, ‘‘Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-power Production or Utilization Facilities*,’’ May 2020, for additional guidance. Note that this DG is associated with the proposed requirements of 10 CFR 50.160(b) which may affect EP requirements for non-LWRs.[8]

16. Aircraft Impact
Construction permit applicants for new nuclear power reactors are required to address the impact of a large commercial aircraft as part of the design. Specifically, 10 CFR 50.150 requires the following:

a. 10 CFR 50.150(a)(1): that each applicant performs a design-specific assessment of the effects on the facility of the impact of a large commercial aircraft. Using realistic analysis, the applicant shall identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions: (1) the reactor core remains cooled, or the containment remains intact; and (2) spent fuel cooling or spent fuel pool integrity is maintained.

b. 10 CFR 50.150(b): that the applicant must include a description of (1) the design features and functional capabilities identified in 10 CFR 50.150 (a)(1), and (2) how the design features and functional capabilities identified in 10 CFR 50.150 (a)(1) meet the assessment requirements in 10 CFR 50.150 (a)(1).

The staff should review the information contained in the CP application and reach conclusions as to whether the applicant has: (1) adequately described design features and functional capabilities in accordance with 10 CFR 50.150(b); and (2) conducted an assessment reasonably formulated to identify design features and functional capabilities to show, with reduced use of operator action, that the facility can withstand the effects of a large commercial aircraft impact.[9] The NRC staff should recognize that the information in the CP application may be based on preliminary design information. Therefore, 10 CFR 50.150 requires applicants to perform the aircraft impact assessment at both licensing stages and include the required information in both applications based on the level of design information available at the time of each application.

[7] Proposed 10 CFR 50.160, “Emergency preparedness for small modular reactors, non-light water reactors, and non-power production or utilization facilities” can be found at 85 FR 28436.
[8] Ibid
[9] Consideration of Aircraft Impacts for New Nuclear Power Reactors, 74 FR 28120 (June 12, 2009).

The staff should consider the review guidance in SRP Section 19.5, "Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts," and RG 1.217, Revision 0, "Guidance for the Assessment of Beyond-Design-Basis Aircraft Impacts," which endorses the guidance in NEI 07-13, Revision 8, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," as an acceptable method for use in satisfying the NRC requirements in 10 CFR 50.150(a) regarding the assessment of aircraft impacts for new nuclear power reactors. When considering the review guidance, the staff should note that the guidance is based on traditional LWR technologies. For non-LWRs, a preapplication discussion with the applicant could aid in addressing the following issues:

- SECY-11-0112, "Staff Assessment of Selected Small Modular Reactor Issues Identified in SECY-10-0034," Enclosure 5, "Aircraft Impact Assessments for Small Modular Reactors," provides considerations for aircraft impact assessments for non-LWRs. This enclosure notes that the four functions identified in 10 CFR 50.150(a)(1) are applicable to LWRs, and may not be applicable to non-LWR reactor designs, or may have to be supplemented by other key functions. When reviewing non-LWR designs, the staff will evaluate the applicability of the acceptance criteria set forth in the aircraft impact rule and the possible need for other criteria. As noted in the statement of considerations for 10 CFR 50.150, if necessary, the staff will issue exemptions and impose supplemental criteria to be used in the aircraft impact assessment for such non-LWR designs.

  SECY-11-0112 also describes areas for additional staff consideration should an application include that ability to produce process heat for industrial use. In such cases the staff should include the impacts resulting from events at the industrial facility associated with the reactor, including aircraft impacts, as part of the external hazards analysis and the siting evaluation.

- SECY-20-0093, "Policy and Licensing Considerations Related to Micro-Reactors," Enclosure 1 includes a discussion of aircraft impact assessments. This enclosure includes the following considerations:

  From a consequence perspective, the staff expects micro-reactors to more closely resemble nonpower reactors than large LWRs. Further, the site footprint of micro-reactors is likely to be substantially smaller than that of the existing power reactor fleet and the new reactors envisioned when the NRC promulgated the aircraft impact rule. Some micro-reactors might also be located underground, which could prevent a large commercial aircraft from striking safety-significant portions of a facility. A holistic risk-informed consideration of design-specific features, including the potential consequences of an aircraft impact, could provide a basis for meeting the underlying purpose of the rule and would be consistent with the Statements of Consideration, which stated that the NRC may need to issue exemptions and impose supplemental criteria for aircraft impact assessments of non-LWRs. Provided a micro-reactor applicant can make a case *for demonstrating compliance with the rule, the staff expects that existing regulatory processes are sufficient to address micro-reactor applications in the near term.*

The staff should note that the aircraft impact rule does not require that the actual aircraft assessment be submitted to the NRC. Therefore, the NRC will address the adequacy of the aircraft impact assessment through an inspection of that assessment that is independent of the licensing review of the application. The licensee is however expected to use the results of the aircraft impact assessment to provide the information identified in SRP 19.5 in its application.

17. Research and Development
The staff should review any identified research and development (R&D) program plans that are designed to resolve any safety questions associated with safety features or components. This review should consider the applicant's plan for research activities including testing of new safety or security features that differ from existing designs for operating reactors, or that use simplified, inherent, passive means to accomplish their safety or security function. The testing should ensure that these new features will perform as predicted, will provide for the collection of sufficient data to validate computer codes, and will show that the effects of system interactions are acceptable.

The staff should ensure that the applicant's commitments to develop sufficient information (through testing or R&D) to support the reliability, availability and performance of safety-related and safety-significant SSCs and human actions modelled in the final PRA (e.g., commitments for items such as fuel testing and analytical code verification and validation) are completed on a schedule to support the staff's review of the final design.

The staff should ensure that the applicant has provided a summary description of preoperational and/or startup testing that is planned for each unique or first-of-a-kind principal design feature that may be included in the facility design or provide information, as applicable, that is sufficient to credit previously performed testing for identical unique or first-of-a-kind design features at other NRC-licensed production facilities.

The staff should conclude that the R&D plans will permit the staff to make the findings required by 10 CFR 50.43(e) (for applications which differ significantly from light-water reactor designs that were licensed before 1997 or use simplified, inherent, passive, or other innovative means to accomplish their safety functions).

18. Fuel qualification
The reactor core and its fuel are generally identified as safety-related due to the direct involvement in performing fundamental safety functions. The information requirements associated with safety-related SSCs are discussed in Section 6, "Safety-Related SSC Criteria and Capabilities." However, there are regulatory requirements, such as fuel design limits, that are attributed-to or identified with fuel performance and its qualification. One of the characteristics of fuel qualification is the need for irradiation data with associated long-time frames to collect that irradiation data. Accordingly, it is anticipated that advanced reactor designs will use existing data (e.g., Advanced Gas Reactor (AGR) program data,

legacy metal fuel data) to support regulatory licensing to some degree. Staff review of fuel qualification at the CP stage should focus on (1) understanding the role of the fuel in the safety case, and (2) determining the adequacy of the plan to provide the evidentiary basis for fuel performance as assumed in the safety case. Sufficient information should be available to support reasonable assurance findings that:

1. The role of the fuel of the safety case is adequately described. This can be addressed by providing fuel performance requirements during (1) normal operation, including the effects of anticipated operational occurrences, and (2) accident conditions. In support of these findings, the staff should seek to understand the safety limits of the fuel and the fuel contribution in the accident source term. Understanding of the safety limits and source term should address uncertainty associated with any limitations on data available at the CP stage and reflected in the analyses discussed in Section 2c "Safety and Accident Analysis Methodologies and Associated Validation" and Section 3b "Discussion of accident source terms" of this paper.
2. The fuel qualification plan is adequate. Staff evaluation 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 utilize lead test specimens. Where legacy data is used, a justification for the applicability of the data to the current application should be provided (e.g., data was collected for a fuel fabricated consistent with the proposed fuel design and irradiated in an applicable environment).

Two NRC documents provide additional guidance in the area of non-LWR FQ:

- NRC draft *white paper "Fuel Qualification for Advanced Reactors (Draft)," dated September 2020* (ML20191A259).
- NRC staff report "Assessment of White Paper Submittals on Fuel Qualification and Mechanistic Source Terms: Next Generation Nuclear Plant", Revision 1, July 2014 (ML14174A845).

19. Regulatory Exemptions

The staff should review the requested exemptions from NRC requirements. The applicant should refer to NRC Staff Draft *White Paper "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors," September 20, 2020* (ML20241A017) for guidance regarding the applicability of NRC regulations to their facility.

20. Environmental Report

The staff should review an applicant's environmental report (ER) as part of the CP application in accordance with 10 CFR 51.50(a). The ER is expected to address the environmental issues described in RG 4.2, "Preparation of Environmental Reports for Nuclear Power Stations," which provides guidance to applicants for the format and content of ERs that are submitted as part of an application for a permit, license, or other authorization to site, construct, and/or operate a new nuclear power plant, or provide a justification for any issues that do not need to be analyzed. Guidance on the review of environmental issues is given in NUREG-1555, "Standard Review Plans for Environmental Reviews for Nuclear Power Plants"

U.S. NUCLEAR REGULATORY COMMISSION

DRAFT REGULATORY GUIDE DG-xxxx

Proposed new Regulatory Guide

Issue Date: xxx 2021
Technical Leads: Joseph Sebrosky
Juan Uribe

This draft staff white paper has been prepared and is being released to support ongoing public discussions. This draft white paper uses Regulatory Guide format because the staff is considering using this format to provide guidance in the near future to support the review of advanced reactor applications. This draft white paper is being issued in parallel with the NRC staff's review of draft industry guidance. The main purpose of this document at this early stage of advanced reactor guidance development is to engage stakeholders on the staff's initial high-level considerations on issues to be considered in such guidance.

This paper has not been subject to NRC management and legal reviews and approvals, and its contents are subject to change and should not be interpreted as official agency positions.

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

A. INTRODUCTION

Purpose

This regulatory guide (RG) provides the U.S. Nuclear Regulatory Commission (NRC) staff's guidance on using a technology-inclusive content of application methodology to inform specific portions of the safety analysis report (SAR) included as part of an advanced reactor license application. Specifically, this RG endorses the methodology described in Nuclear Energy Institute (NEI) 21-xx, "XYZ" (Ref. x), as one acceptable process for use when developing portions of an application for an advanced reactor construction permit, operating license, combined license, manufacturing license,

This regulatory guide is being issued in draft form to involve the public in the early stages of the development of a regulatory position in this area. It has not received final staff review or approval and does not represent an official NRC final staff position. Public comments are being solicited on this draft guide (including any implementation schedule) and its associated regulatory analysis or value/impact statement. Comments should be accompanied by appropriate supporting data. Written comments may be submitted through the government-wide Web site http://www.regulations.gov. Alternatively, written comments by be submitted to the Rules, Announcements, and Directives Branch, Office of Administration, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; or faxed to (301) 492-3446. Copies of comments received may be examined at the NRC's Public Document Room, 11555 Rockville Pike, Rockville, MD. Comments will be most helpful if received by [insert date – 60 days from issuance].

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://nrcweb.nrc.gov/reading-rm/doc-collections/reg-guides/. 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. MLxyz. The regulatory analysis may be found in ADAMS under Accession No. MLxyz.

standard design approval (SDA), or design certification under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic Licensing of Production and Utilization Facilities” (Ref. x), and 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants” (Ref. x). It is anticipated that this guidance will be updated to use for reviews of advanced nuclear reactor license and permit applications submitted under 10 CFR Part 53, “Licensing and Regulation of Advanced Nuclear Reactors,” once that regulation is final.

NEI 21-xx describes an approach to develop the scope and content of an application by implementing the licensing modernization project (LMP) methodology described in NEI 18-04, Revision 1, “Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development” (Ref. x) as endorsed by the NRC in RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications For Licenses, Certifications, and Approvals for Non-Light-Water Reactors” (Ref. x). The methodology in NEI 18-04 provides a systematic, risk-informed and technology neutral process for developing key inputs into the content of applications, to improve understanding of the safety and risk significance of system designs and their relationship to safety evaluations for a variety of non-light water reactor (LWR) designs. Even though the guidance described in NEI 18-04 is intended for non-LWRs, the NRC staff believes that the content and methodology described is also an acceptable approach to develop an application for the other categories of advanced reactors.

In this RG, the term “advanced reactor” is used in the context of the Nuclear Energy Innovation and Modernization Act (NEIMA). NEIMA included a definition for “advanced nuclear reactor” that was further refined by the Energy Act of 2020. The definition of advanced nuclear reactor found in the Energy Act of 2020 includes:

> (1) ADVANCED NUCLEAR REACTOR. —The term ‘advanced nuclear reactor’ means—
> (A) a nuclear fission reactor, including a prototype plant (as defined in sections 50.2 and 52.1 of title 10, Code of Federal Regulations (or successor regulations)), with significant improvements compared to reactors operating on the date of enactment of the Energy Act of 2020, including improvements such as—
> (i) additional inherent safety features;
> (ii) lower waste yields;
> (iii) improved fuel and material performance;
> (iv) increased tolerance to loss of fuel cooling;
> (v) enhanced reliability or improved resilience;
> (vi) increased proliferation resistance;
> (vii) increased thermal efficiency;
> (viii) reduced consumption of cooling water and other environmental impacts;
> (ix) the ability to integrate into electric applications and nonelectric applications;
> (x) modular sizes to allow for deployment that corresponds with the demand for electricity or process heat; and
> (xi) operational flexibility to respond to changes in demand for electricity or process heat and to complement integration with intermittent renewable energy or energy storage; and
> (B) a fusion reactor.

In SECY 20-0032, “Rulemaking Plan On “Risk-Informed, Technology-Inclusive Regulatory Framework for Advanced Reactors (Rin-3150-Ak31; Nrc-2019-0062)” (Ref. x), the staff further clarified its interpretation of the advanced reactors described in NEIMA to include LWR small modular reactors (SMRs), non-LWRs, and fusion reactor designs.

Although the technology-inclusive methodology described in NEI 21-xx provides a common approach to identifying and describing the scope and level of detail for the fundamental safety functions of a design, The applicant or the licensee is still responsible for demonstrating compliance with all applicable regulations and may request exemptions as appropriate. The staff issued a white paper to provide guidance on which regulations are applicable to non-LWRs in September of 2020, titled "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors," (Ref. ML20241A017). The staff supplemented this white paper in a document dated February 2021 (Ref. ML21049A098). The September 2020, white paper as supplemented by the February 2021 document describe which regulations are generally applicable to non-LWR applications for construction permits and operating licenses under 10 CFR Part 50 and standard design certifications, combined licenses, and standard design approvals under 10 CFR Part 52. The staff is in the process of revising the applicability of regulations white paper. The staff intends to include the content of this white paper as Appendix D the ARCAP roadmap ISG.
In addition, the staff notes that the applicability of specific technical requirements in NRC regulations or the need to define additional technical requirements for a particular design arising from the safety assessments will be made on a case-by-case basis for advanced reactors.

Applicability

This RG applies to nuclear power reactor designers, applicants, and licensees of advanced reactors[1] (non--LWR and SMR designs) applying for permits, licenses, certifications, and approvals under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. x), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. x). It is envisioned that the review approach described in this RG will also support the technology-inclusive, risk-informed and performance-based application content and level of detail expected in a future application submitted under the proposed Title 10 of the Code of Federal Regulations (10 CFR) Part 53, "Licensing and Regulation of Advanced Nuclear Reactors," which is currently being developed.

Applicable Regulations

- 10 CFR Part 50 provides regulations for licensing production and utilization facilities.
  - 10 CFR 50.34, "Contents of applications; technical information," describes the minimum information required for (a) preliminary safety analysis reports supporting applications for a construction permit and (b) final safety analysis reports supporting applications for operating licenses.
- 10 CFR Part 52 governs the issuance of early site permits, standard design certifications (DCs), combined licenses (COLs), standard design approvals (SDAs), and manufacturing licenses (MLs) for nuclear power facilities.
  - 10 CFR 52.47, "Contents of applications; technical information," describes the information to be included in final safety analysis reports supporting applications for standard DCs.

[1] Certain elements of this RG may also be applicable to Fusion reactors, as appropriate. However, the staff notes that options for the regulatory treatment of fusion reactors are currently being considered by the NRC staff which may result in the development of fusion-specific guidance.

- 10 CFR 52.79, “Contents of applications; technical information in final safety analysis report,” describes the information to be included in final safety analysis reports supporting COLs.

- 10 CFR 52.137, “Contents of applications; technical information,” describes the information to be included in final safety analysis reports supporting SDAs.

- 10 CFR 52.157, “Contents of applications; technical information in final safety analysis report,” describes the information to be included in final safety analysis reports supporting MLs.

Related Guidance, Communications, and Policy Statements

- “Policy Statement on the Regulation of Advanced Reactors” (Volume 73 of the *Federal Register*, page 60612, October 14, 2008) (Ref. x), establishes the Commission’s expectations related to advanced reactor designs to protect the environment and public health and safety and promote the common defense and security with respect to advanced reactors.

- RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition)” (Ref. x), provides detailed guidance to the writers of safety analysis reports to allow for the standardization of information the NRC requires for granting construction permits and operating licenses.

- RG 1.232, “Guidance for Developing Principal Design Criteria for Non-Light Water Reactors” (Ref. x), describes the NRC’s guidance on how the general design criteria in 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants.” This guidance may be used by non-LWR reactor designers, applicants, and licensees to develop principal design criteria for any non-LWR designs, as required by the applicable NRC regulations for nuclear power plants. The RG also describes the NRC’s guidance for modifying and supplementing the general design criteria to develop principal design criteria that address two types of non-LWR technologies: sodium cooled fast reactors and modular high temperature gas-cooled reactors (MHTGRs).

Purpose of Regulatory Guides

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

Paperwork Reduction Act

This RG provides voluntary guidance for implementing the mandatory information collections in 10 CFR Parts 50 and 52 that are subject to the Paperwork Reduction Act of 1995 (44 U.S.C. 3501 et. seq.). These information collections were approved by the Office of Management and Budget (OMB), approval numbers 3150-0011 and 3150-0151. Send comments regarding this information collection to the Information Services Branch (T-6A10M), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, or by e-mail to Infocollects.Resource@nrc.gov, and to the OMB reviewer at: OMB Office of Information and Regulatory Affairs (3150-0011 and 3150-0151), Attn: Desk Officer for the Nuclear Regulatory Commission, 725 17th Street, NW Washington, DC20503; e- mail: oira_submission@omb.eop.gov.

Public Protection Notification

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

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[ToC will be updated in the final stages of the document]

B. DISCUSSION

Reason for Issuance

This RG provides the U.S. Nuclear Regulatory Commission (NRC) staff's guidance on using a technology-inclusive content of application methodology to inform specific portions of the safety analysis report (SAR) included as part of an advanced reactor license application. Specifically, this RG endorses the methodology described in Nuclear Energy Institute (NEI) 21-xx, "XYZ" (Ref. x), as one acceptable process for use when developing portions of an application for an advanced reactor construction permit, operating license, combined license, manufacturing license, standard design approval (SDA), or design certification under Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. x), and 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. x).

Background

As the NRC prepares to review and regulate a new generation of advanced reactors, the staff has previously recognized the need to establish, and the benefits of having, a flexible regulatory framework. The NRC described 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," (Ref. x). The staff then developed "NRC Non-Light Water Reactor Near Term Implementation Action Plans" (Ref. x), and "NRC Non-Light Water Reactor Mid-Term and Long-Term Implementation Action Plans" (Ref. x), to identify specific activities that the NRC will conduct in the near-term, mid-term, and long term timeframes. Similarly, the Commission encouraged the use of a performance based technology inclusive licensing framework for SMRs in SRM - COMGBJ-10-0004/COMGEA-10-0001, "Use of Risk Insights to Enhance Safety Focus of Small Modular Reactor Reviews," and SRM – SECY -11-0024, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews".

To ensure review readiness, a key element of this new and flexible regulatory framework is to standardize the development of content within an advanced reactor application to promote uniformity among applicants. A standardized content of application for advanced reactors also ensures review consistency and predictability from NRC staff, and presents a well-defined base from which to evaluate proposed changes in the scope and requirements of reviews. The development of applications for NRC licenses, permits, certifications, and approvals is a major undertaking, in that an applicant must provide sufficient information to support the agency's safety findings. The needed information and level of detail will vary according to whether an application is for a construction permit, design approval, design certification, operating license, combined license, or other action.

The NRC staff has had success with a standard content of application methodology for large-LWRs. The NRC's efforts to standardize the format and content of applications for LWRs are reflected in RG 1.70, issued in the 1970s, and RG 1.206, issued in 2007 and revised in 2018. Guidance documents, such as NUREG 0800 and numerous other documents on specific technical areas, address the suggested scope and level of detail for applications.

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

- The Advanced Reactor Content of Application Project (ARCAP), and
- The Technology-Inclusive Content of Application Project (TICAP).

The ARCAP is an NRC-led activity, and is intended to provide guidance for a complete advanced reactor application that supports 10 CFR Part 50, 10 CFR Part 52, and the ongoing 10 CFR Part 53 rulemaking effort. As a result, ARCAP is broad and encompasses several industry-led, and NRC-led guidance documents aimed at ensuring a consistent approach to the development of each application document. A complete advanced reactor application is expected to include, among other things, a SAR, a Quality Assurance plan, a Fire Protection program, Emergency and Physical Security plans, etc.

The TICAP is an industry-led activity, and is focused on providing guidance on the appropriate scope and depth of information related to the specific portions of the SAR that describe the fundamental safety functions of the design, and details the affirmative safety case for each applicant consistent with the LMP approach. TICAP's focus on those measures needed to address risks posed by non-LWR and SMR technologies will help an applicant provide sufficient information on the design and programmatic controls, while avoiding an excessive level of detail on less important parts of a plant. The specific portions of the SAR applicable to the scope of NEI 21-xx are described below in more detail. Based on the limited scope of the TICAP guidance, TICAP's scope is encompassed by and supplemented by the ARCAP guidance. The ARCAP will describe the guidance for the specific areas of the SAR that are outside the scope of the LMP process (i.e., not covered by TICAP) such as site information, and information consistent with use of the American Society of Mechanical Engineers (ASME) Section III, Division 5 construction codes.

As a result of extensive TICAP/ARCAP public interactions with industry and external stakeholders, the proposed development of the SAR for an advanced reactor application is based on a 12-chapter approach. In contrast, the SAR approach for large-LWRs described in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," (Ref. x) is based on a 19-chapter approach. For an advanced reactor application consistent with ARCAP/TICAP and the methodology described in this RG, the 12 chapters are as follows:

- Chapter 1 - General Plant Information, Site Description, and Overview of the Safety Case
- Chapter 2 - Generic Analyses
- Chapter 3 - License Basis Event Analysis
- Chapter 4 - Integrated Evaluations
- Chapter 5- Safety Functions, Design Criteria, and SSC Categorization
- Chapter 6 - Safety Related SSC Criteria and Capabilities
- Chapter 7 - NSRST SSC Criteria and Capabilities
- Chapter 8 - Plant Programs
- Chapter 9 - Routine Plant Radioactive Effluents, Plant Contamination, and Solid Waste
- Chapter 10 - Control of Occupational Dose
- Chapter 11 - Organization
- Chapter 12 - Initial Startup Programs.

Based on the SAR structure described above, the staff notes that TICAP's scope as described in NEI 21-xx is only applicable to the LMP-related portions contained in the first 8 chapters[2]. Figure 1 below illustrates the nexus between ARCAP, TICAP, and other guidance in relation to an advanced reactor application.

[2] Pre-application engagement is highly encouraged for applicants that plan to use the methodology described in NEI-21 xx but rely on a different SAR structure than the 12-chapter approach described in this RG. Similarly, applicants not using the LMP approach described in NEI 21-xx but leveraging the 12-chapter SAR approach should engage the NRC staff early to optimize application reviews. On October 2020, the staff issued a white paper related to the importance of pre-application activities consistent with the Commission's advanced reactor policy statement (ref. x).

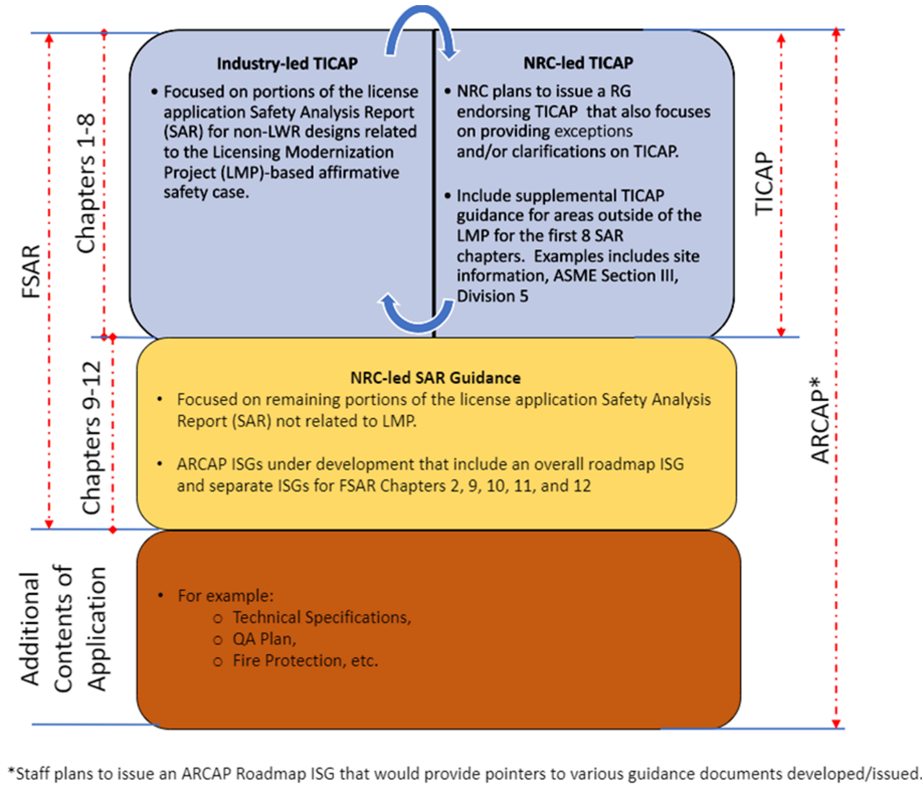

Figure 1: Nexus between ARCAP, TICAP, and the content of an application.

Documents Endorsed in this Guide

Upon completion of the industry-led TICAP efforts, the results of the project were documented as guidance in NEI 21-xx, and submitted to NRC for review and endorsement. As a result, the purpose of this RG is twofold:

1. To endorse certain sections of NEI 21-xx which describe one acceptable approach for determining the scope and level of detail for the development of structured application content associated with the first 8 chapters of the SAR. The methodology in NEI 21-xx follows the LMP guidance, and systematically describes the selection of licensing-basis events (LBEs); classification and special treatments of structures, systems, and components (SSCs); the assessment of defense in depth (DID) features and supporting information. When applicable, this RG will also describe any additional clarifications, exceptions, points of emphasis, and/or further details relevant to the specific sections discussed in NEI 21-xx and endorsed by this RG.

2. To describe additional information outside the scope of LMP and NEI 21-xx that NRC staff has determined is also relevant and would expect to be included as part of the application content related to the first 8 chapters.

Based on the above, this RG endorses Sections {x,y, and z}of NEI 21-xx as one acceptable process for use when developing content for portions of the NRC license application SAR for non-LWR and SMR designs in a manner consistent with 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." Additional details for each chapter will be described in their corresponding section.

The NRC endorsement of the aforementioned sections does not imply the NRC's endorsement of the references cited in the endorsed sections of NEI 21-xx or to references in the endorsed sections of NEI 21-xx to other (unendorsed) sections of NEI 21-xx. The NRC has not necessarily reviewed and approved the guidance provided in these references, except where specifically noted in this regulatory guide.

In summary, the guidance in NEI 21-XX is focused on developing the portions of the SAR containing material addressed in NEI 18-04, and it will help ensure completeness of information submitted to NRC while avoiding unnecessary burden on the applicant and rightsizing the content of application commensurate with the complexity of the design being reviewed. This guidance provides a standardized content development process designed to facilitate efficient preparation by the applicant, review by the regulator, and maintenance by the licensee. The content formulation should optimize the type and level of detail of information provided, based on the complexity of the design's safety case and the nexus between elements of the design and public health and safety.

Harmonization with International Standards

As described in the 2010 IAEA Integrated Regulatory Review Service (IRRS) mission report, the NRC has agreed to review international standards and, when practical, harmonize NRC regulations and guides with the appropriate international standards. During the development of this RG, the NRC staff did not identify any international standards related to this guide.

C. STAFF REGULATORY GUIDANCE

The guidance on the SAR content scope and level of detail described in this RG is based on the appropriate level of design-specific information that should be provided in an application to the NRC to demonstrate that the design's safety case meets the regulatory standards for adequate protection of public health and safety. To accommodate an effective and efficient technology inclusive content guidance while ensuring the underlying intent of the current content requirements is met, this guidance is formulated to describe an LMP-based affirmative safety case. Pre-application engagement between applicants and the NRC is highly encouraged to optimize resources and review schedule, especially for non-LMP based applications.

The following sections describe the NRC's endorsement (with clarifications or exceptions, when applicable) of the corresponding sections described in guidance document NEI-21 xx. In general, NEI 21-xx is structured to present the overall safety case first and then provide the specific supporting design and operating details in subsequent chapters. The staff notes that the methods, approaches, or data described in the regulatory position(s) discussed below are not requirements.

1. <u>General Plant and Site Description, And Overview of The Safety Case</u>

The information in this chapter should allow the reviewer to obtain a basic understanding of the overall facility, such as the type of permit, license, certification or approval requested, the number of plant 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 a site, and how they relate to the affirmative safety case. Examples of information related to site description include geological and demographic, seismological, hydrological, and meteorological characteristics of the site and the surrounding vicinity.

Chapter 1 of NEI 21-xx, Rev. x (specifically, Sections 1.1 to 1.4) provides an acceptable method for licensees to follow and develop baseline information related to the plant description, site description, the affirmative safety case based on the LMP methodology, and a summary of reference of source materials, respectively.

For reference, the affirmative safety case is defined as a collection of scientific, technical, administrative and managerial evidence which documents the basis that the performance objectives of the technology-inclusive fundamental safety functions (FSFs) are met by a design during design specific Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), Beyond Design Basis Events (BDBEs), and Design Basis Accidents (DBAs) by:

- Identifying design specific safety functions that are adequately performed by design specific SSCs and
- Establishing design specific features (programmatic (e.g., inspections) or physical (e.g., redundancy)) to provide reasonable assurance that credited SSC functions are reliably performed.

1st Regulatory Position

NEI 21-xx, Chapter 1 provides an acceptable method for developing information related to the plant description, site description, the affirmative safety case based on the LMP methodology, and a summary of reference of source materials. However, the applicant or the licensee is still responsible for demonstrating compliance with all applicable regulations and may request exemptions as appropriate. The staff issued a white paper to provide guidance on which regulations are applicable to non-LWRs in September of 2020, titled "Analysis of Applicability of NRC Regulations for Non-Light Water Reactors," (Ref. ML20241A017). The staff supplemented this white paper in a document dated February 2021 (Ref. ML21049A098). The September 2020, white paper as supplemented by the February 2021 document describe which regulations are generally applicable to non-LWR applications for construction permits and operating licenses under 10 CFR Part 50 and standard design certifications, combined licenses, and standard design approvals under 10 CFR Part 52. The staff is in the process of revising the applicability of regulations white paper. The staff intends to include the content of this white paper as Appendix D the ARCAP roadmap ISG.

2nd Regulatory Position – Construction Permit Information

NEI 21-xx, Section xxx provides an acceptable method for developing portions of a construction permit application in accordance with 10 CFR Part 50 requirements. However, for advanced reactor applicants pursuing a construction permit (CP) application under 10 CFR Part 50 and using an alternative risk-informed performance-based approach (such as LMP), additional information not related to the LMP-based affirmative safety case should be provided. Specifically, the additional information is related to the minimum information necessary in a CP application for the staff to issue a CP under 10 CFR 50.35(a) when the applicant has not supplied all of the technical information required to complete the application (i.e., 50.34(a)) and support the issuance of a CP which approves all proposed design features (i.e., obtains finality for the design). The staff notes that the additional CP information described in this RG is consistent with the first 8 Chapters of the SAR. The guidance described in Appendix A of this RG contains guidance on one acceptable approach in scope and level of detail for applicants to provide the additional relevant CP information for advanced reactor applications related to the first 8 chapters of the SAR.

3rd Regulatory Position – Supplemental Information
In addition to the material identified in NEI 21-xx, Chapter 1 of the SAR should also address the following issues:

a. Identify the applicability of Generic Safety Issues, Unresolved Safety Issues and Three Mile Island action items to the design and their proposed resolution.

b. Identify the RGs applicable to the design and any proposed exceptions.

c. Identify the consensus design codes and standards (ASME, ANSI, IEEE, etc.) used in the design along with what SSCs to which they apply. This includes, as appropriate, reference to ASME B&PV Code Section III, Division 5, "High Temperature Reactors."

2. Generic Analyses

An important part of the design process for reactor designs is the identification of events that could challenge key safety functions and layers of defense against the release of radioactive materials. Therefore, a key part of the review of an advanced reactor application is the selection of licensing basis events (LBEs). The LBEs are described as event sequences such as anticipated operational occurrences (AOOs), design-basis events (DBEs), or beyond-design-basis events (BDBEs). The primary determinate for categorizing events in each of these categories is the estimated frequency of the event sequence. Figure 3-2 of NEI 18-04 provides additional information on the selection and evaluation of LBE's.

Chapter 2 of NEI 21-xx, Rev. x (specifically, Sections 2.1 to 2.3) provides an acceptable method for licensees to follow and develop baseline information related to the probabilistic risk assessment (overview of the PRA), source-term analysis, and design-basis accidents (DBAs) analytical methods.

4th Regulatory Position
NEI 21-xx, Chapter 2 provides an acceptable method for developing information related to the probabilistic risk assessment (overview of the PRA), source-term analysis, and design-basis accidents analytical methods.

5th Regulatory Position – Site Information
In addition to the site information described in Chapter 2 of NEI 21-xx, additional information not developed using the LMP process should be provided. The purpose of this information is to demonstrate compliance with 10 CFR 100, Subpart B, and the relevant parts of 10 CFR 50 and 52 that discuss site related issues, and to describe the site characteristics used in the design and safety analysis where (i) a design basis external hazard level must be specified for each system, structure, or component (SSCs) designed to withstand this hazard with no adverse impact on their capability to perform their required safety function (RSF) or (ii) an SSC is relied upon to establish the adequacy of defense-in-depth and must be designed with special treatment to withstand a given hazard. The guidance described in draft Interim Staff Guidance (ISG) "Site Information" (ADAMS Accession No. ML20316A013) contains guidance on one acceptable approach in scope and level of detail for applicants to provide relevant site information.

6th Regulatory Position - Generic Analyses
Certain analyses are common to a number of LBE analyses. This chapter of the guidance provides information regarding how to document those analyses in an application. The scope of content, the level of detail, and the structure of the application guidance regarding this topic are acceptable, with the following exceptions and clarifications:

a. Section **3.2.1** (PRA) states that the summary PRA information in this chapter is not considered in change control evaluations even though it provides the key PRA findings. The staff's position is that a change control program should be provided to address changes to PRA and that a summary of this process be included in the application.

b. Section **3.2.1** (PRA) states that the PRA information in SAR is only "for information." Some of the information such as commitments to the non-LWR standards, certain assumptions, reliability targets, etc. should be part of the licensing basis as such information is relied on by the staff for the licensing decision (or to make the reasonable assurance of safety finding for the NRC) for advanced reactor applications using the LMP.

c. Other generic analysis that should be provided include baseline operating parameters; a description of systems, components, and materials performance under normal operating, anticipated transient, and accident conditions.

3. <u>Licensing basis Events</u>

After the identification of LBEs, the information in this chapter should describe the systematic and reproducible process and methodology used to select the LBEs, and the specific analysis and evaluation of the selected LBEs against the proposed design. The analysis in this section is primarily described in terms of event sequences comprised 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. This chapter should also include information on the process used to group and condense the substantial number of event sequences considered in the PRA into sequence families that are used to define the AOOs, DBEs, and BDBEs. It is important to note that the term "event sequence" is used in lieu of the term "accident sequence" used in LWR PRA standards because the scope of the LBEs includes AOOs and initiating events with no adverse impacts on public safety.

Chapter 3 of NEI 21-xx, Rev. x (specifically, Sections 3.1 to 3.6) provides an acceptable method for licensees to follow and develop baseline information related to the LBE selection methodology, and summary of LBEs (AOOs, DBEs, BDBEs and DBAs).

<u>*7th Regulatory Position*</u>
NEI 21-xx, Chapter 3 provides an acceptable method for developing information related to the LBE selection methodology, and summary of LBEs (AOOs, DBEs, BDBEs and DBAs).

<u>*8th Regulatory Position – Supplemental Information*</u>
In addition to the material identified in NEI 21-xx, Chapter 3, the SAR should also include a discussion of the following:

1. Aircraft Impact Assessment (10 CFR 50.150) – The objective of the aircraft impact rule is to require nuclear power plant designers to rigorously assess their designs to identify design features and functional capabilities that could provide additional inherent protection to withstand the effects of an aircraft impact. The NRC expects this rule to result in new nuclear power reactor facilities that are inherently more robust with regard to an aircraft impact than if they were designed in the absence of the aircraft impact rule. The rule provides an enhanced level of protection beyond that which is provided by the existing adequate protection requirements applicable to currently operating power reactors. The following Regulatory Guide (RG) provides guidance regarding implementation this regulation:

    - RG 1.217, "Guidance for The Assessment of Beyond-Design-Basis Aircraft Impacts," describes a method that the staff of the NRC considers acceptable for use in satisfying the regulations at 10 CFR 50.150, regarding the consideration of aircraft impacts for new nuclear power reactors. In particular, this RG endorses the methodologies described in the industry guidance document, Nuclear Energy Institute (NEI) 07-13, "Methodology for Performing Aircraft Impact Assessments for New Plant Designs," Revision 8, dated April 2011.

2. Mitigation of Beyond-Design-Basis External Events from Natural Phenomena (circumstances associated with loss of large areas of the plant due to explosions or fire) (10 CFR 50.155) – One of the primary lessons learned from the accident at the Fukushima Dai-ichi nuclear power plant was the significance of the challenge presented by a loss of multiple safety-related systems following the occurrence of a beyond-design-basis external event (BDBEE). 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, external flooding, etc.) beyond those accounted for in the design basis, while unlikely, could present challenges to nuclear power plants. The following RGs provide guidance regarding implementation this regulation:

    - RG 1.226, "Flexible Mitigation Strategies for Beyond-Design-Basis Events," identifies methods and procedures the staff of the NRC considers acceptable for nuclear power reactor applicants and licensees to demonstrate compliance with NRC regulations covering planning and preparedness for beyond-design basis events as required by 10 CFR 50.155, "Mitigation of beyond design-basis events." This RG endorses, with clarifications, the methods and procedures promulgated by the Nuclear Energy Institute (NEI) in technical document NEI 12-06, "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide," Revision 4 (NEI 12-06, Revision 4) dated December 2016 as a process the NRC considers acceptable for meeting, in part, the regulations in 10 CFR 50.155. Additionally, this RG provides guidance for meeting the regulations in 10 CFR 50.155 that are in areas that are not covered in NEI 12-06.

    - RG 1.227, "Wide-Range Spent Fuel Pool Level Instrumentation," identifies methods and procedures the staff of the NRC considers acceptable for demonstrating compliance with NRC regulations to provide 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, "Mitigation of beyond-design-basis events" (10 CFR 50.155). This RG endorses, with exceptions and clarifications, the methods and procedures promulgated by the Nuclear Energy Institute (NEI) in document NEI 12-02, "Industry Guidance for Compliance with NRC Order EA-12-051, 'To Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation'," Revision 1 (NEI 12-02) dated August 2012 as a process the NRC staff considers acceptable for meeting certain regulations in 10 CFR 50.155.

    - Draft Regulatory Guide DG-1319 (Proposed New Regulatory Guide 1.228), "Integrated Response Capabilities for Beyond-Design-Basis Events," identifies methods and procedures the staff of the NRC considers acceptable for nuclear power reactor applicants and licensees to demonstrate compliance with 10 CFR 50.155, and Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Section VII, "Communications and Staffing Requirements for the Mitigation of Beyond Design Basis Events." This RG endorses, with clarifications, the methods and procedures promulgated by the Nuclear Energy Institute (NEI) in the following documents as methods the NRC staff considers acceptable for meeting portions of the regulations in 10 CFR 50.155 and 10 CFR Part 50, Appendix E, Section VII:

- NEI 12-01, "Guidelines for Assessing Beyond-Design-Basis Accident Response Staffing and Communication Capabilities," Revision 0, dated May 2012.

- NEI 13-06, "Enhancements to Emergency Response Capabilities for Beyond-Design-Basis Events and Severe Accidents," Revision 0, dated September 2014, and NEI 14-01, "Emergency Response Procedures and Guidelines for Beyond-Design-Basis Events and Severe Accidents." Revision 0 dated September 2014.

4. Integrated Evaluations

The information in this chapter should describe the integrated risk of all LBEs against the plant, and evaluated against three cumulative risk targets:

1. The total mean frequency of exceeding a site boundary dose of 100 millirem (mrem) from all LBEs should not exceed 1/plant-year. The value of 100 mrem is selected from the annual cumulative exposure limits in 10 CFR 20.
2. The average individual risk of early fatality within 1 mile of the exclusion area boundary (EAB) from all LBEs based on mean estimates of frequencies and consequences shall not exceed 5×10-7/plant-year to ensure that the NRC safety goal quantitative health objective (QHO) for early fatality risk is met.
3. 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 shall not exceed 2×10-6/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.

Key information in this chapter should be to identify design features that are responsible for preventing and mitigating radiological releases and for meeting the integrated risk criteria, including defense-in-depth (DID). This evaluation leads to performance requirements and design criteria that are developed within the process of the SSC classification. This chapter should also describe information that conveys the evaluated SSC margins against the total mean frequency of exceeding a site boundary dose of 100 mrem in order to establish baseline margins between the frequencies and consequences of individual LBEs against the frequency-consequence curve described in Figure 3-1 of NEI 18-04, Rev. 1

Chapter 4 of NEI 21-xx, Rev. x (specifically, Sections 4.1 and 4.2) provides an acceptable method for licensees to follow and develop baseline information related to the Integrated Evaluations, which include the overall plant risk performance summary, and identify design features that are responsible for preventing and mitigating radiological releases and for meeting the integrated risk criteria, including defense-in-depth.

9th Regulatory Position
NEI 21-xx, Chapter 4 provides an acceptable method for developing information related to the Integrated Evaluations, which include the overall plant risk performance summary, and identify design features that are responsible for preventing and mitigating radiological releases and for meeting the integrated risk criteria, including defense-in-depth.

5. Safety Functions, Design Criteria, and Systems, Structures, and Components Classification

As part of the LMP process, LBEs are generally defined in terms of successes and failures of SSCs that perform safety functions and are modeled in the probabilistic risk-assessment (PRA). Therefore, the PRA safety functions (PSFs) are those functions responsible for the prevention and mitigation of an unplanned radiological release from any source within the plant.

The information in this chapter should describe the approach for designating SSC safety functions and classifications in accordance with the PSFs. For SSCs, information should include a description of the required safety function(s) (RSFs), required functional design criteria (RFDC), principal design criteria (PDC), safety classification of safety-related, and non-safety related with special treatment (NSRST) SSCs, and the complementary design criteria. Definitions for these terms are described in Section 6 of NEI 18-04, Rev. 1 "Glossary of Terms." The information in this chapter should also identify potential technical concerns related to SSC safety classification, and the derivation of requirements necessary to support PSFs in the prevention and mitigation of LBEs that are modeled in the PRA.

Chapter 5 of NEI 21-xx, Rev. x (specifically, Sections 5.1 to 5.4) provides an acceptable method for licensees to follow and develop baseline information related to the safety classification of SSCs, which includes information about RSFs, RFDC, PDCs, and NSRST.

10th Regulatory Position
NEI 21-xx, Chapter 5 provides an acceptable method for developing information related to the safety classification of SSCs, which includes information about RSFs, RFDC, PDCs, and NSRST.

11th Regulatory Position – Supplemental Information – Fuel Qualification
(Note this is preliminary language and will be updated as appropriate based on staff guidance that is under development. Two NRC documents provide additional guidance in the area of non-LWR fuel qualification: 1) NRC draft white paper "Fuel Qualification for Advanced Reactors (Draft)," dated September 2020 (ML20191A259), and 2) NRC staff report "Assessment of White Paper Submittals on Fuel Qualification and Mechanistic Source Terms: Next Generation Nuclear Plant", Revision 1, July 2014 (ML14174A845)).

In addition to the material identified in NEI 21-xx, Chapter 5 of the SAR should also address fuel qualification. The reactor core and its fuel are generally identified as safety-related due to the direct involvement in performing fundamental safety functions. The information requirements associated with safety-related SSCs are discussed in Section 6, "Safety-Related SSC Criteria and Capabilities." However, there are regulatory requirements, such as fuel design limits, that are attributed-to or identified with fuel performance and its qualification. One of the characteristics of fuel qualification is the need for irradiation data with associated long-time frames to collect that irradiation data. Accordingly, it is anticipated that advanced reactor designs will use existing data (e.g., Advanced Gas Reactor (AGR) program data, legacy metal fuel data) to support regulatory licensing to some degree. The fuel qualification discussion should focus on (1) understanding the role of the fuel in the safety case, and (2) determining the adequacy of the plan to provide the evidentiary basis for fuel performance as assumed in the safety case. Sufficient information should be available to support reasonable assurance findings that:

1) The role of the fuel in the safety case is adequately described. This can be addressed by providing fuel performance requirements during (1) normal operation, including the effects of anticipated operational occurrences, and (2) accident conditions. In support of these findings, sufficient information should be provided such that the safety limits of the fuel and the fuel contribution in the accident source term are clearly identified. Understanding of the safety limits and source term should address uncertainty associated with any limitations on data available and reflected in the analyses discussed in Section 2 "Safety and Accident Analysis Methodologies and Associated Validation" and Section 3 "Discussion of accident source terms" of NEI 21-xx.

2) The fuel qualification plan is adequate. 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 utilize lead test specimens. Where legacy data is used, a justification for the applicability of the data to the current application should be provided (e.g., data was collected for

a fuel fabricated consistent with the proposed fuel design and irradiated in an applicable environment).

6. Safety-Related Systems, Structures, and Components Criteria and Capabilities

The information in this chapter should leverage the analysis performed for the safety related SSCs in Chapter 5 of NEI 21-xx and describe further detail into the criteria and capabilities of all safety related SSCs that are part of the affirmative safety case.

For SSCs classified as SR, the information in this chapter should address the design criteria referred to as Safety-Related Design Criteria (SRDC). The SRDC are derived from the RFDC, which in turn are developed from the RSFs determined in the LBE selection process described in Chapters 2 and 3 of NEI 21-xx.

Chapter 6 of NEI 21-xx, Rev. x (specifically, Sections 6.1 to 6.3) provides an acceptable method for licensees to follow and develop baseline information related to the design requirements for SSCs, special treatment requirements for SSCs, and system descriptions for safety-related SSCs.

12th Regulatory Position
NEI 21-xx, Chapter 6 provides an acceptable method for developing information related to the design requirements for SSCs, special treatment requirements for SSCs, and system descriptions for safety-related SSCs.

13th Regulatory Position – Supplemental Information
In addition to the material identified in NEI 21-xx, Chapter 6 of the SAR should also address the following:

1. If there are instrumentation and control systems that are identified as safety related then Design Review Guide (DRG), "Instrumentation and Controls for Non-Light-Water Reactor (non-LWR) Reviews," (ADAMS under Accession No. ML21011A140) provides additional guidance for content and review of this material.

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

The information in this chapter should describe the regulatory 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.: not SR SCCs), but are relied upon to perform risk-significant functions. Risk-significant SSCs are those that perform functions that prevent or mitigate any LBE from exceeding the frequency-consequence target or make significant contributions to the cumulative risk metrics selected for evaluating the total risk from all analyzed LBEs.

For clarity the term "special treatment" is derived from NRC regulations and Nuclear Energy Institute (NEI) guidelines in the implementation of 10 CFR 50.69. In Regulatory Guide 1.201, the following definition of special treatment is provided:

> "…special treatment refers to those requirements that provide increased assurance beyond normal industrial practices that structures, systems, and components (SSCs) perform their design-basis functions."

Chapter 7 of NEI 21-xx, Rev. x (specifically, Sections 7.1 and 7.2) provides an acceptable method for licensees to follow and develop baseline information related to the special treatment requirements for NSRST SSCs at the site, and NSRST SSCs descriptions and capabilities. Additional information can be found in Table 4-1 of NEI 18-04.

14th Regulatory Position
NEI 21-xx, Chapter 7 provides an acceptable method for developing information related to the special treatment requirements for NSRST SSCs at the site, and NSRST SSCs descriptions and capabilities.

15th Regulatory Position – Supplemental Information
In addition to the material identified in NEI 21-xx, Chapter 6 of the SAR should also address the following:

a. If there are instrumentation and control systems that are identified as safety related then Design Review Guide (DRG), "Instrumentation and Controls for Non-Light-Water Reactor (non-LWR) Reviews," (ADAMS under Accession No. ML21011A140) provides additional guidance for content and review of this material.

8. Plant Programs

The information in this chapter should provide information on those plant programs relied upon to provide special treatment to SR and NSRST SSCs that are part of the affirmative safety case. The information should provide an overview of the special treatment programs, addressing the purpose, scope, and performance objectives as well as applicability to SSCs. The information for the programs should provide reasonable assurance that 1) reliability and performance targets are met, and 2) safety-significant uncertainties are addressed. Program areas could include human factors, quality assurance, startup testing, and equipment qualification, among others.

Chapter 8 of NEI 21-xx, Rev. x (specifically, Sections X and Y) provides an acceptable method for licensees to follow and develop baseline information related to those plant programs relied upon to provide special treatment to SR and NSRST SSCs that are part of the affirmative safety case.

16th Regulatory Position
NEI 21-xx, Chapter 8 provides an acceptable method for developing information related to those plant programs relied upon to provide special treatment to SR and NSRST SSCs that are part of the affirmative safety case.

17th Regulatory Position – Supplemental Information
In addition to the material identified in NEI 21-xx, Chapter 8 of the SAR should also address the following:

a. A discussion of SR SSCs and their treatment should be provided in Chapter 8 of the SAR. The term "special treatment" is used in a manner consistent with NRC regulations and Nuclear Energy Institute (NEI) guidelines in the implementation of 10 CFR 50.69. In Regulatory Guide 1.201, the following definition of special treatment is provided:

> "…special treatment refers to those requirements that provide increased assurance beyond normal industrial practices that structures, systems, and components (SSCs) perform their design-basis functions."

All safety-significant SSCs are subject to special treatment requirements. Chapter 8 of the SAR should describe special treatment requirements applicable to each SR SSC. These requirements should include specific performance requirements to provide adequate assurance that the SSCs will be capable of performing their RSFs with significant margins and with appropriate degrees of reliability. These include numerical targets for SSC reliability and availability, design margins for performance of the RSFs, and monitoring of performance against these targets with appropriate corrective actions when targets are not fully realized. Another consideration in the setting of SSC performance requirements is the need to assure that the results of the plant capability DID evaluation described in Chapter 4 and 5 of the application (in accordance with NEI 21-xx) are achieved not just in the design, but in the as-built and as-operated and maintained plant throughout the life of the plant.

b. Associated testing/validation for SR SSCs
Special treatment requirements for SR SSCs may include the performance of routine testing and validation of SSC performance capability. Describe, as applicable, the special treatment requirements from NEI 18-04, Table 4-1, on a case-by-case basis and in the context of the SSC functions in the prevention and mitigation of applicable LBEs. These special treatment items for SR SSC may include the following:

- Equipment qualification - Essentially the same as for existing reactors for SR SSCs, 10 CFR 50.49
- Materials qualification
- Pre-service and risk-informed in-service inspections - See Regulatory Guide 1.178
- Pre-service and in-service testing - In–service testing needs to be integrated with Reliability Assurance Program
- Surveillance testing - Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met (i.e., demonstrate the ability to perform the safety function).

c. All NSRST SSCs are subject to special treatment requirements. This Chapter should describe special treatment requirements applicable to each NSRST SSC. These requirements should include specific performance requirements to provide adequate assurance that the SSCs will be capable of performing their RSFs with significant margins and with appropriate degrees of reliability. These include numerical targets for SSC reliability and availability, design margins for performance of the RSFs, and monitoring of performance against these targets with appropriate corrective actions when targets are not fully realized. Another consideration in the setting of SSC performance requirements is the need to assure that the results of the plant capability DID evaluation described in Chapter 4 and 5 of the application (in accordance with NEI 21-xx) are achieved not just in the design, but in the as-built and as-operated and maintained plant throughout the life of the plant.

d. Associated testing/validation for NSRST SSCs
Special treatment requirements for NSRST SSCs may include the performance of routine testing and validation of SSC performance capability. Describe, as applicable, the special treatment requirements from NEI 18-04, Table 4-1, on a case-by-case basis and in the context of the SSC functions in the prevention and mitigation of applicable LBEs. These special treatment items for NSRST SSCs may include the following:

- Reliability assurance targets
- Seismic qualification
- Pre-service and risk-informed in-service inspections - See Regulatory Guide 1.178

- Pre-service and in-service testing - In–service testing needs to be integrated with Reliability Assurance Program
- Surveillance testing - Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions for operation will be met (i.e., demonstrate the ability to perform the safety function).

D. IMPLEMENTATION

The purpose of this section is to provide information on how advanced reactor applicants and licensees[3] may use this guide and information regarding the NRC's plans for using this RG. In addition, it describes how the NRC staff complies with 10 CFR 50.109, "Backfitting," and any applicable finality provisions in 10 CFR Part 52.

Use by Applicants and Licensees

Advanced reactor applicants and licensees may voluntarily[4] use the guidance in this document to demonstrate compliance with the underlying NRC regulations. Methods or solutions that differ from those described in this RG may be deemed acceptable if they provide sufficient basis and information for the NRC staff to verify that the proposed alternative demonstrates compliance with the appropriate NRC regulations.

Advanced reactor licensees may use the information in this RG for actions that do not require NRC review and approval such as changes to a facility design under 10 CFR 50.59, "Changes, Tests, and Experiments." Non-LWR licensees may also use the information in this RG or applicable parts to resolve regulatory or inspection issues.

Use by NRC Staff

The NRC staff does not intend or approve any imposition or backfitting of the guidance in this RG. Because this guidance applies only to advanced reactors, and not to power reactors that are large LWRs, the NRC staff does not expect any existing licensee to use or commit to using the guidance in this RG. The NRC staff does not expect or plan to request licensees to voluntarily adopt this RG to resolve a generic regulatory issue. The NRC staff does not expect or plan to initiate NRC regulatory action which would require the use of this RG. Examples of such unplanned NRC regulatory actions include issuance of an order requiring the use of the RG, requests for information under 10 CFR 50.54(f) as to whether a licensee intends to commit to use of this RG, generic communication, or promulgation of a rule requiring the use of this RG without further backfit consideration.

During regulatory discussions on plant specific operational issues, the staff may discuss with licensees various actions consistent with staff positions in this RG, as one acceptable means of meeting the underlying NRC regulatory requirement. Such discussions would not ordinarily be considered backfitting even if prior versions of this RG are part of the licensing basis of the facility. However, unless this RG is part of the facility license, the staff may not represent to the licensee that the licensee's failure to comply with the positions in this RG constitutes a violation.

If a licensee voluntarily seeks a license amendment or change and (1) the NRC staff's consideration of the request involves a regulatory issue directly relevant to this new or revised RG and

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3 In this section, "licensees" refers to licensees of nuclear power plants under 10 CFR Parts 50 and 52; and the term "applicants," refers to applicants for licenses and permits for (or relating to) nuclear power plants under 10 CFR Parts 50 and 52, and applicants for standard design approvals and standard design certifications under 10 CFR Part 52.

4 In this section, "voluntary" and "voluntarily" means that the licensee is seeking the action of its own accord, without the force of a legally binding requirement or an NRC representation of further licensing or enforcement action.

(2) the specific subject matter of this RG is an essential consideration in the staff's determination of the acceptability of the licensee's request, then the staff may request that the licensee either follow the guidance in this RG or provide an equivalent alternative process that demonstrates compliance with the underlying NRC regulatory requirements. This is not considered backfitting as defined in 10 CFR 50.109(a)(1) or a violation of any of the issue finality provisions in 10 CFR Part 52. Additionally, an existing applicant may be required to comply with new rules, orders, or guidance if 10 CFR 50.109(a)(3) applies.

If a licensee believes that the NRC is either using this RG or requesting or requiring the licensee to implement the methods or processes in this RG in a manner inconsistent with the discussion in this Implementation section, then the licensee may file a backfit appeal with the NRC in accordance with the guidance in NRC Management Directive 8.4, "Management of Facility-Specific Backfitting and Information Collection" (Ref. 32), and in NUREG-1409, "Backfitting Guidelines," (Ref. 34).

ACRONYMS/ABBREVIATIONS – To Be Updated

AOO	anticipated operational occurrence
BDBE	beyond-design-basis event
CFR	*Code of Federal Regulations*
COL	combined license
DC	design certification
DBA	design-basis accident
DBE	design-basis event
DBEHL	design-basis external hazard level
DG	draft regulatory guide
DID	defense in depth
DOE	U.S. Department of Energy
F-C	frequency-consequence
IAEA	International Atomic Energy Agency
IDP	integrated decision panel
ITAAC	inspections, tests, analyses, and acceptance criteria
ITP	Initial Test Program
LBE	licensing-basis event
LMP	Licensing Modernization Project
LWR	light-water reactor
MHTGR	modular high-temperature gas-cooled reactor
ML	manufacturing license
mrem	millirem
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
NSRST	nonsafety-related with special treatment
OMB	Office of Management and Budget
PRA	probabilistic risk assessment
PRISM	Power Reactor Innovative Small Module
RFDC	required functional design criteria
RG	regulatory guide
RTNSS	regulatory treatment of nonsafety systems
SDA	standard design approval
SR	safety related
SRM	staff requirements memorandum
SSC	structure, system, and component
TS	technical specification
U.S.C.	*United States Code*

REFERENCES – To Be Developed [5]

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

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

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, Phone: 202-739-800, Fax 202-785-4019.

Appendix A
Construction Permit Application Guidance

Detailed Advanced Reactor Construction Permit Guidance

This guidance is intended for CP applications involving advanced reactors following the licensing modernization project process. The guidance is based on an application using a risk-informed performance-based approach, such as the advanced reactor content of application project (ARCAP) whose purpose is to develop technology-inclusive, risk-informed and performance-based application guidance. The ARCAP, documented in *ISG-XXX, "Advanced Reactor Content of Application Interim Staff Guidance,"* is broad and encompasses the industry-led technology-inclusive content of application project (TICAP). This CP guidance references applicable guidance developed through the ARCAP/TICAP activities as well as guidance derived from separate ongoing regulatory activities (e.g., security and emergency planning rulemaking), as necessary.

The TICAP guidance that is being developed in parallel with the guidance found in this document is based on the Licensing Modernization Project (LMP) described in NEI 18-04, Revision 1, "Risk-Informed Performance-Based Guidance for Non-Light Water Reactor Licensing Basis Development," as endorsed by the NRC in Regulatory Guide (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." Several vendors have indicated that they plan to implement the LMP to develop the licensing basis for their applications. As such, processes from the LMP and initial guidance referencing TICAP and ARCAP draft documents are referenced throughout this document.

The ARCAP guidance is currently under development and is intended to be used in conjunction with the guidance in this document for the review of a non-LWR CP application. Because ARCAP/TICAP is in its early stages this document italicizes NRC guidance and industry standards that are under development that are not yet formally endorsed. These italics will be removed in future revisions to the document as the ARCAP/TICAP guidance and other NRC guidance and Industry standards to reflect the appropriate endorsed guidance.

However, applicants are not required to utilize the TICAP/LMP approach and may instead use another methodology (e.g., traditional deterministic approach, maximum hypothetical accident[6]) to analyze non-LWR performance and develop a licensing basis. The TICAP/LMP process forms the basis for this guidance although in some areas the guidance provides additional considerations for acceptably addressing a specific topic when a TICAP/LMP approach is not used. As noted above applicants are encouraged to use the preapplication process to optimize reviews, which is especially important if an applicant intends to use a process other than the LMP to develop their licensing basis. Regardless, the review guidance in this document is limited in scope. The NRC staff should continue to consult other established guidance documents, as applicable, to complete reviews of non-LWR applications.

This guidance addresses the minimum information necessary in a CP application for the NRC staff to issue a CP under Title 10 of the *Code of Federal Regulations* (10 CFR) 50.35(a) when the applicant has not supplied all of the technical information required to complete the application (i.e., 10 CFR 50.34(a)) and support the issuance of a CP which approves all proposed design features (i.e., obtains finality for the

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[6] In this context, "maximum hypothetical accident" refers to a conservatively assessed, deterministic accident with consequences that bound the full spectrum of accident conditions for the plant and is not necessarily a credible event.

design). When making its safety finding regarding the issuance of a CP under 10 CFR 50.35(a), the NRC staff should make the determination that the application:

(1) Describes the proposed design of the facility, including, but not limited to,
   a. the principal architectural and engineering criteria for the design, and
   b. the major features or components incorporated therein for the protection of the health and safety of the public.

(2) Describes safety features or components, if any, which require research and development program necessary to resolve any safety questions associated with such features or components.

(3) Provides commitments that such safety questions will be satisfactorily resolved at or before the latest date stated in the application for completion of construction of the proposed facility, and

(4) Describes the site criteria contained in 10 CFR Part 100 and based on that criteria concludes that the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

Where an applicant desires design finality regarding a specific topic, the NRC staff should review that the application has provided sufficient information about the topic at a level of detail that is expected at the operating license (OL) stage. The guidance that follows is limited to the first 8 chapters of the preliminary safety analysis report (PSAR) consistent with the scope and methodology described in Nuclear Energy Institute (NEI) 21-xx, "XYZ." For CP guidance outside the first 8 chapters of the PSAR, refer to *draft* ARCAP ISG.

Specific Topic Guidance

1. General Plant and Site Description
   The NRC staff should review application content to ensure that the following information is included:
   a. Overview of technology (size of the reactor and planned commercial application of the design—power production, industrial application, etc.), including references to previous experience with similar designs and technology.

   b. General plant and site characteristics including:
      i. The specific number, type, lifetime, and thermal power level of the facilities, or range of possible facilities, for which the site may be used.
      ii. General description of the important plant design and operational features in sufficient detail to allow the reviewer to understand how the plant operates in normal and off-normal conditions, including refueling. The description should include the major plant structures, systems, and components (SSCs) and relied upon to meet the regulations. The important characteristics (coolant, moderator, fuel design, neutron spectra, materials, etc.) of the design. Drawings and other material as necessary to understand the design.
      iii. A description of how the design accomplishes the fundamental safety functions of controlling reactivity, heat removal, and radionuclide retention, including spent fuel storage and cooling, should be provided.
      iv. The Principal Design Criteria (PDCs) applicable to the design (for additional guidance on selecting PDCs, refer to RG 1.232 "Guidance for Developing Principal Design Criteria for Non-Light Water Reactors."

v. A summary of the approach used in conducting the safety analysis, including Licensing Basis Events (LBEs) including Design Basis Accidents (DBAs), safety classification of SSCs and their performance requirements and special treatments, adequacy of defense-in-depth (DID) and the overall acceptance criteria used.
vi. Overview of the analytical codes and analysis methods used.
vii. The location and boundaries of the site.
viii. The proposed general location of each major structure on the site.

c. Novel design features – provide a description of novel design features (such as passive systems, inherent safety features, or simplified control features) that may be used in safety-related or safety-significant SSCs. Topics to be considered beyond the reactor system include unique features such as seismic isolators, novel digital instrumentation and control systems, security features, or novel approaches to programs.

d. Identify the applicability of Generic Safety Issues, Unresolved Safety Issues and Three Mile Island action items to the design and their proposed resolution.

e. Identify the RGs applicable to the design and any proposed exceptions.

f. Identify the consensus design codes and standards (ASME, ANSI, IEEE, etc.) used in the design along with what SSCs they apply to.

2. Generic Analysis
   a. Source Terms
   The NRC staff should review the source term methodology used by the applicant to include the validation and verification of the associated engineering computer programs. The source term development needs to include radiological source terms for accident analysis, routine effluents, radwaste system design, shielding design and equipment qualification. The NRC staff should consider the guidance and references found in SECY-16-0072, "Accident Source terms and Siting for Small Modular Reactors and Non-Light Water Reactors" (ML15309A319) for additional information regarding expected CP application content in this area.

   b. PRA
   The NRC staff should review the description and results of the applicant's PRA described in a CP application. The plant design and the associated PRA at the CP application stage are less mature relative to the Operating License and, accordingly, are considered to be preliminary. Therefore, the description of the PRA is a high-level overview or summary that covers topics such as the methodology, scope, and acceptability of the PRA. When assessing the acceptability of the PRA, the NRC staff should consider any self-assessment, use of the non-LWR PRA standard (ASME/ANS RA-S-1.4-2021) including any exceptions, and/or peer review performed by the applicant commensurate with the plant design and PRA development stage. The description of PRA should also discuss how insights gained from the PRA have been, and will be, used during the design and construction of the plant. The NRC staff should examine the methods used or to be used to conduct a thorough and systematic search for initiating events (such as the use of master logic diagrams, heat balance fault trees, process hazards analysis, failure modes and effects analysis, operating experience reviews, etc.). The results of PRA should summarize the key outputs of the PRA including risk-significant LBEs, SSCs and human actions as well as other risk insights such as those on

defense-in-depth. The results should also discuss the uncertainty analysis and sensitivity analysis performed. The NRC staff reviews the planned further development of the PRA and the use of its results to help resolve any safety questions associated with the major features or components identified in a CP application.

In order for the NRC staff to conclude that the PRA is of sufficient scope and technical adequacy to support a CP application, the staff needs to be assured that:

- The PRA description addresses the methodology used, includes a discussion regarding initiating events, includes key outputs and risk insights, and describes further plans for PRA development and use.
- The methodology is generally consistent with either consensus industry standards or good industry practices.
- The search for initiating events was complete given the level of design completeness.
- The PRA results were properly derived.
- Insights identified were incorporated into preliminary designs.

The reviewer should first understand the context in which the PRA is being used, which includes the description and results of the PRA. The description of PRA should include the key PRA assumptions. To assess the quality of the PRA for the decision-making in support of the application, it is expected that the applicants conform with the guidance provided in Section 2.1, "Probabilistic Risk Assessment," of NEI 21-xxx regarding the content of a SAR related to PRA.

In addition, the frequencies and probabilities should be appropriately estimated; and the engineering analyses, assumptions, and approximations used in developing the PRA model be appropriate and should demonstrate the robustness with respect to the uncertainties in the assessment.

The NRC staff should make evaluation findings that the PRA has been performed in such a way that the PRA results are reasonable based on the level of maturity of the design, and information provided in the SAR is reasonable and sufficient to support the findings.

c. Safety and Accident Analysis
The staff should review the safety and accident analysis used by the applicant to support findings associated with 10 CFR 50.34(a)(4). This review should consider that the requirement under 10 CFR 50.43(e)(1)(iii), that sufficient data exist on the safety-features of the design to assess the analytical tools used for safety analysis, is not applicable to a CP. Accordingly, evaluation of the safety margins using approved evaluation models is not required to support a CP. However, preliminary analyses should be available to support reasonable assurance findings that:

1. The design will be able to provide sufficient margins of safety during normal operations and transient conditions.
2. The applicant has identified the structures, systems, and components necessary for the prevention of accidents and the mitigation of the consequences of accidents.
3. The applicant has demonstrated an understanding of the uncertainty associated with the performance of structures, systems, and components necessary for the prevention of accidents and the mitigation of the consequences of accidents.

It is noted that items above are closely related (e.g., 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) that safety features or components which require research and development have been described and that there will be conducted a research and development program reasonably designed to resolve any safety questions associated with such features or components (see Section 17 on Research & Development). Additionally, the review of the safety analysis should consider the identification of licensing basis events (see Section 3 on licensing basis events).

d. Site Information
The NRC staff should review the site information in the application. Guidance regarding specific information content for this section can be found in draft *ARCAP ISG, "Site Information*," (for applications using the LMP approach) *and [forthcoming] Staff Requirements Memorandum (SRM) to SECY-20-0045, "Population-Related Siting Considerations for Advanced Reactors,"* for guidance regarding population distribution. The relevant topics areas are:
   i. Site Characteristics and Site Parameters (Overview)
   ii. Geography and Demography
      (1) Site Location and Description
      (2) Exclusion Area Authority and Control
      (3) Population Distribution
   iii. Nearby Industrial, Transportation, and Military Facilities
   iv. Regional Climatology, Local Meteorology, and Atmospheric Dispersion
   v. Hydrological Description
      (1) Floods
      (2) Flooding Protection
      (3) Groundwater
   vi. Geology, Seismology, and Geotechnical Engineering
      (1) Geologic Hazards
      (2) Vibratory Ground Motion
      (3) Surface Deformation
      (4) Stability of Subsurface Materials and Foundations
      (5) Stability of Slopes
   vii. Summary of Design Basis External Hazards

3. Licensing Basis Events
The NRC staff should review the process described in the application for selection of LBEs and classification and treatment of SSCs. One acceptable approach is described in RG 1.233, which classifies LBEs as either Anticipated Operational Occurrences (AOOs), Design Basis Events (DBEs), Beyond Design Basis Events (BDBEs), or DBAs. DBAs are selected from the set of DBEs. Other risk-informed approaches will need to be reviewed, evaluated, and determined acceptable by the staff. Regardless of the approach described for addressing LBEs and classification and treatment of SSCs, the staff review should ensure that the application adequately describes the analysis of the radiological consequences of accidents to show compliance with 10 CFR 50.34(a)(1), to include the following:

a. Discussion of selected DBAs. The NRC staff should ensure that the spectrum of DBAs includes those DBAs that present the greatest challenge with respect to calculated fission product releases.

b. Discussion of accident source terms. The NRC staff should consider the following:
   i. The identification of radionuclide release mechanisms from fuel, the associated limits, and the contribution to source term are or will be supported by experimental data that cover the needed range of applicability.
   ii. The performance of fission product barriers credited to prevent and/or inhibit the release of radionuclides are or will be supported by existing or planned experimental data that cover the needed range of applicability.

   The NRC staff should evaluate the applicant's use of bounding assumptions and conservative modeling to account for the uncertainty in final design details. For review of mechanistic source terms (if provided), additional information on development of accident source terms can be found in [INL paper] "Technology-Inclusive Determination of Mechanistic Source Terms for Offsite Dose-Related Assessments for Advanced Nuclear Reactor Facilities," (ML20192A250). The staff should consider SECY-16-0012, "Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors," for guidance on mechanistic source terms.

c. Discussion of the major SSCs of the facility that are intended to mitigate the radiological consequences of a DBA with a description of how the three fundamental safety functions are accomplished for each DBA. Major SSCs of the facility include those that may affect the performance of barriers that restrict or limit the transport of radioactive materials from the fuel to the public (i.e., that bear significantly on the acceptability of the site under the radiological consequence evaluation factors identified in 10 CFR 50.34(a)(1)). The staff's review should include identification of the design basis for the SSCs (e.g., codes and standards to be followed, seismic categories, etc.) as well as the SSC fission product removal mechanisms. This includes natural fission product removal processes or for unique features of the design that may require additional information from the applicant to fully explain the process being credited, the amount of removal being credited (specifically decontamination factors or coefficients and timing), basis for the proposed values and inputs to the dose analysis calculation, and the justification for assuming the removal process is applicable to the design of the plant for the duration of the event

d. Discussion of the characteristics of fission product releases from the proposed site to the environment including the rates of fission product release, the isotopic quantities and the chemical forms of fission products released to the environment.

e. Discussion of the meteorological characteristics of the proposed site used in the accident analysis including the site-specific short-term atmospheric dispersion (χ/Q) values determined by the applicant.

f. Discussion of the analysis methods, assumptions and results for the total calculated radiological consequence dose at the exclusion area boundary (EAB), the outer boundary of the low population zone (LPZ) and control room (if required, e.g., operator actions are relied upon for safety-significant functions) from the DBAs. The uncertainty analyses in the mechanistic source terms and radiological doses should be reviewed as part of the evaluation of conservative assumptions used in this analysis. The plant design features intended to mitigate the radiological consequences of accidents, 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 (calculated at the upper 95th percentile of consequences) fall within the following exposure acceptance criteria specified in 10 CFR 50.34(a)(1)(ii)(D):

i. 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 total effective dose equivalent (TEDE), and
ii. 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.

The NRC staff should consider performing an independent confirmatory radiological consequence analysis using pertinent information in the application to assess whether the proposed site meets the radiological consequence evaluation factors identified in 10 CFR 50.34(a)(1).

4. Integrated Evaluations
   a. Evaluation of Integrated Plant Risk
   Integrated individual risks of site boundary dose and early and latent health effects should be reviewed over the range of LBEs analyzed. The analysis method and assumptions should be reviewed for consistency with NRC practice. Considerations could include:

   - was off-site evacuation in accordance with the facility's EP plan assumed?
   - was medical treatment for those members of the public exposed assumed?
   - what latent fatality risk coefficient was used
   - what segment of the population [average healthy individual or something else] does the risk coefficient represent, etc.?

   The integrated risk evaluation should be reviewed against three cumulative risk targets:
   i. The total mean value frequency of exceeding a site boundary dose of 100 mrem from all LBEs should not exceed 1/plant-year. The value of 100 mrem is selected from the annual exposure limits in 10 CFR Part 20.
   ii. The average individual risk of early fatality within 1 mile of the EAB shall not exceed a mean value of 5×10^{-7}/plant-year to ensure that the NRC safety goal Quantitative Health Objective (QHO) for early fatality risk is met.
   iii. The average individual risk of latent cancer fatalities within 10 miles of the EAB shall not exceed a mean value of 2×10^{-6}/plant-year to ensure that the NRC safety goal QHO for latent cancer fatality risk is met.

   b. Defense-in-Depth
   DID is a design approach to account for uncertainties in equipment and human performance. It can result in redundant, diverse and independent measures to accomplish safety functions and ensure that safety is not dependent upon a single SSC or human action. For applications that use a risk-informed performance-based approach, the staff should expect the DID information to address the systematic assessment methodology endorsed by RG 1.233 and

document preliminary integrated decision-making process panel (IDPP) decisions according to NEI 18-04, Revision 1.

The staff should ensure that the applicant has provided necessary commitments to establish DID adequacy. Commitments to implement the DID evaluation processes in RG 1.233 should be adequate. Alternately, the staff should ensure that the applicant's DID process involves incorporating DID into design features, operating and emergency procedures, and other programmatic elements to ensure performance requirements are maintained throughout the life of the plant. For applicants that choose not to use the RG 1.233 endorsed approach, the applicant will need to explain its approach to DID and include in the application a description regarding how DID is addressed.

5. Safety Functions, Design Criteria, and SSC Categorization
    a. Principal Design Criteria
    The NRC staff should review the PDCs proposed in the application. The NRC staff expects prospective non-LWR applicants will review the general design criteria (GDCs) pertaining to LWRs provided in Appendix A to 10 CFR Part 50 and the guidance in RG 1.232 to develop their PDCs and ensure that necessary safety functions and SSCs are covered under the selected PDCs. The staff should determine that the PDCs were appropriately developed. As part of this process, the staff should evaluate the acceptability the safety functions (referred to as the required safety functions (RSFs) in the LMP process) that must be fulfilled to keep the DBEs within the dose and integrated risk targets. Required Functional Design Criteria (RFDC) are then derived from the RSFs. The staff should ensure that the RFDCs are defined to capture design-specific criteria that may be used to supplement or modify the applicable GDCs or Advanced Reactor Design Criteria in the formulation of PDCs.

    b. Safety-Related (SR) SSCs
    The NRC staff should review the list of the SR SSCs identified through the LBE analysis. The staff should ensure that for each SR SSC, the basis for such classification is indicated in a traceable manner.

    c. Complementary Design Criteria
    The NRC staff should review the complementary design criteria (CDCs) proposed in the application. The staff should determine that the CDCs were appropriately developed. As part of this process, the staff should evaluate the acceptability the risk significant functions that must be fulfilled to address DID adequacy. The NRC staff should ensure that necessary risk significant safety functions and other safety functions for adequate DID are covered under the selected CDC.

    d. Non-Safety-Related with Special Treatment (NSRST) SSCs
    The NRC staff should review the list of the NSRST SSCs identified through the LBE analysis. The staff should ensure that for each NSRST SSC, the basis for such classification is indicated in a traceable manner.

    e. SSC Categorization Process
    The NRC staff should review the SSC categorization process described in the application. NRC accepted guidance for SSC categorization includes RG 1.233 which endorses the

methodology in NEI 18-04, RG 1.201, "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance," and NEI-00-04, "10 CFR 50.69 SSC Categorization Guideline."

6. Safety-Related SSC Criteria and Capabilities
   Refer to NEI 18-04 for a definition of SR SSCs. The NRC staff should review the SR design criteria and special treatment requirements for each SR SSC described in the application. Information should be provided for each SR SSC to support a determination that the SSC will meet its reliability and performance targets as credited in the PRA. [Specifically, the staff should review information for each SR SSC including:
   - Design requirements and applicable codes and standards used in the design of the SSC.
   - The RSF of the SSC, its RFDCs and its relationship to the PDCs.

   The NRC staff should ensure that the application describes how the SR SSCs that are credited in the fulfillment of RSFs are capable to perform their RSFs with a high degree of confidence in response to any Design Basis External Hazard Levels (DBEHLs).

   The NRC staff should ensure that commitments are provided to describe SR SSC reliability and capability performance requirements, performance of testing and validation of SSC performance capability, operability/availability requirements, special treatment requirements, and any required support functions at the operating license stage.

7. Non-Safety Related with Special Treatment (NSRST) SSC Criteria and Capabilities
   Refer to NEI 18-04 for a definition of NSRST SSCs. The NRC staff should review the design criteria and special treatment requirements for each NSTST SSC described in the application. Information should be provided for each NSRST SSC to support a determination that the SSC will meet its reliability and performance targets as credited in the PRA. Specifically, the staff should review information for each NSRST SSC including:

   - Design requirements and applicable codes and standards used in the design.
   - The risk significant functions and functions required for defense-in-depth of the SSC, and its relation to the PDCs (In TICAP these PDCs are called CDCs).

   The staff should ensure that the application describes how the NSRST SSCs are capable of performing their risk-significant functions or functions that are necessary for defense-in-depth adequacy with a high degree of confidence in response to any internal hazard (e.g., internal floods, internal fires, pipe whip, spatial placement, etc.) or DBEHLs.

   The staff should ensure that commitments are provided to describe NSRST SSC reliability and capability performance requirements, performance of testing and validation of SSC performance capability, availability requirements, special treatment requirements, and any required support functions at the OL stage.

8. Plant Programs
   The NRC staff should review the application for commitments to develop programs needed to implement the special treatments and meet reliability and performance targets for SR SSCs and NSRST SSCs. Such program areas may include in-service testing, maintenance, human factors, training, and reliability assurance.