This draft staff white paper has been prepared and is being released to support ongoing public discussions.

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.

NRC STAFF WHITE PAPER

SAFETY REVIEW OF POWER REACTOR CONSTRUCTION PERMIT APPLICATIONS

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC, or Commission) staff is providing this draft white paper to facilitate discussion of the safety review of light-water reactor (LWR) and non-LWR construction permit (CP) applications for power reactors.

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".[1]
- 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

[1] 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.

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 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.

Lessons Learned from Recently Issued CPs

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.

Existing Light-Water Reactor Safety Review Guidance

The SRP in NUREG-0800 provides guidance to assure quality and predictability in the staff's safety review of various licensing actions, including an LWR CP application. The guidance in the SRP, along with the additional guidance provided later in this document, provides the reviewer with an acceptable approach to meet the applicable requirements in the regulations for LWR applications. For advanced reactor applications following the ARCAP format, the reader should consider the guidance in Appendix C of this document.

Implementation of the acceptance criteria contained in the SRP and the additional guidance in this document provides assurance that an LWR design will comply with the Commission's regulations and provide adequate protection of the public health and safety. Applications for licenses under 10 CFR Parts 50 and 52 typically follow the structure of the SRP to efficiently support the staff's safety review of the applications. Except when an applicant proposes an alternative method or standard for complying with the regulations applicable to the licensing action, the staff will use the methods described in the SRP and this document to evaluate the application's conformance with the Commission's regulations. In cases where an applicant proposes to use an alternative approach or standard in its application, the staff will evaluate the information to ascertain whether the alternative approach demonstrates compliance with the requirements, including maintaining sufficient design margins.

Recent updates to the SRP focused on guidance to support the review of COL applications submitted under 10 CFR Part 52. Many SRP sections retained separate guidance for the review of a CP application while other SRP sections consolidated that guidance in the review procedures for applications submitted under 10 CFR Part 52. The special topics section of this white paper provides additional information on potential CP application submissions, specific CP safety review areas needing clarity, and applicability of guidance resulting from the activities to a develop technology-inclusive framework for both non-LWRs and LWRs.

In addition to the SRP, RGs 1.70 and 1.206 provide guidance on the format, content, and level of detail for license applications submitted under 10 CFR Parts 50 and 52. Although the guidance in RG 1.70 dates from the 1970s and the guidance in RG 1.206 applies to a COL application, the information in these RGs support a CP application structure consistent with the SRP, help to ensure completeness of information in the application, and provide insights on what information in the application would support the staff's safety review and evaluation. Although the RGs provide insights, the staff should use the SRP to guide its review as superseded or supplemented by new or revised regulations, other regulatory guidance, staff analyses of previous applications, and other published staff positions, being mindful of Commission policy in Management Directive 8.4 (Ref. 11) on using, in appropriate circumstances, the same reasoned decision-making process as used for forward fits. In addition, the staff should approach its review consistent with the expectations for new reactor reviews documented in the August 20, 2018, memorandum from Frederick Brown (Ref. 3), and apply the principles of good regulation discussed in the October 15, 2019, memorandum from Ho Nieh (Ref. 4).

The approach to reviewing a CP application is intended to be different from the more recent COL application reviews in which an applicant provided all technical information on the final design of the facility to support the Commission findings for issuance of a COL under 10 CFR Part 52. As discussed in the original proposed Part 52 rule (53 FR 32060, August 23, 1988), the licensing process in 10 CFR Part 50 "was structured to allow licensing decisions to be made while design work was still in progress and to focus on case-specific reviews of individual plant and site considerations. Construction permits were commonly issued with the understanding that open safety issues would be addressed and resolved during construction, and that issuance of a construction permit did not constitute Commission approval of any design feature. Consequently, the operating license review was very broad in scope."

Therefore, the staff's review and evaluation of the proposed design of a facility provided in a CP application is the first stage of a continuing review of the design, construction, and operating features described in the applicant's PSAR. The plant design and operating features may be preliminary for the initiation of construction, with NRC evaluation of the final design, including any design changes made during construction, occurring during the review of the subsequent OL 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.

Clarifications for Light Water Reactors CP Applications Following the SRP Structure

The LWR application guidance in RGs 1.70 and 1.206 provides insights on the structure, information, and level of detail in the application needed to support the staff's review. These guidance documents are consistent with the SRP structure and review procedures found in NUREG-0800 to assure quality and predictability in the staff's safety review.

The reviewer should approach the CP application consistent with the SRP guidance in NUREG-0800, the insights in Appendix C of this document that may be applicable to an LWR CP application, and the following clarifications. These clarifications address a subset of topics that are reviewed in the CP application. The following is not intended to be inclusive of all topics expected in a CP application.

Siting

The staff should review the CP application information on the facility and the physical characteristics of the proposed site (including the geological, seismological, hydrological, and meteorological characteristics of the site and vicinity), in conjunction with present and projected population distribution, land use, site activities and controls, and potential man-related hazards. The staff's review of these topics should determine how these site characteristics have influenced plant design and operating criteria and to show the adequacy of the site characteristics from a safety viewpoint. The SRP provides guidance for reviewing these technical areas and includes the requirements of 10 CFR Part 100, "Reactor Site Criteria," related to the development of the security and emergency plans. It is expected that the applicant completely characterizes the site selected for construction. Also, the application should include a commitment that, if an unexpected effect is detected during construction, the OL applicant will provide an acceptable analysis of the problem and a plan of action to eliminate or significantly reduce the harmful effects or damage.

Radiological Consequence Analyses

In reviewing a CP application with preliminary design information, the staff should consider the applicant's use of bounding assumptions to account for uncertainty in final design and the potential for different methods presented in the FSAR accompanying the OL application. The staff should approach the review of safety and siting analyses commensurate with the specificity of the design details and safety assessment in the application with a focus on the major safety features and components in the design that support site suitability. In a CP review for a preliminary design, the staff should not need final design details for systems, structures, and

components unless the applicant is requesting approval of specific design features in its CP application.

Transient and Accident Analyses

The preliminary analysis and evaluation of a nuclear power plant should include analyses of the response of the plant to postulated disturbances in process variables and to postulated malfunctions or failures of equipment. Such safety analyses provide a significant contribution to the selection of limiting conditions for operation, limiting safety system settings, and design specifications for components and systems from the standpoint of public health and safety. These analyses are a focal point of the Commission's construction permit reviews of facilities to support a finding that the proposed facility can be constructed and operated without undue risk to the health and safety of the public as required by 10 CFR 50.34 and 50.35.

It is essential that all credible accidents be considered and evaluated during the CP application stage. The accident analyses should include the effects of anticipated process disturbances and postulated component failures to determine their consequences and to evaluate the capability of the design to control or accommodate such failures. The situations analyzed should include anticipated operational occurrences and postulated accidents.

The review of transients and accident analyses requires an evaluation of analytical methods, inputs, and results of analyses. In most cases, analytical methods are not documented in the application; instead, an applicant may refer to a vendor topical report. Examples of such methods for LWR designs include DNB (departure from nucleate boiling) correlation development, subchannel analysis, system transient analysis, analysis of RIA (reactivity-initiated accidents), and LOCA (loss-of-coolant accident) analysis. For those cases where applicants use techniques previously considered and approved by the staff, the reviewer verifies the previously approved method is applicable and stipulated limitations and conditions are satisfied. However, if new methods are involved, a review of topical reports and other information which describe the method of analysis is performed. Such a review generally includes vendor model description, data correlations and empirical relationships, solution techniques, summary of computer codes if involved, sample problems, experimental verification, and comparative calculations.

The reviewer should ensure the preliminary analysis and evaluation has considered a sufficiently broad spectrum of initiating events; ensure the initiating events are categorized by type and frequency of occurrence to confirm the selected events are limiting; and verify that the results of selected transients and accidents satisfy pertinent figures of merit and acceptance criteria. The reviewer verifies that the applicant systematically analyzed and evaluated the limiting events in each category using a detailed quantitative analysis. At a minimum a reviewer should ensure the preliminary safety analysis report includes all the information required by 10 CFR 50.34, with a focus on:

- Evaluations of the design and SSC performance resulting from operation of the facility;
- Determination of the margins of safety during normal operations and transient conditions anticipated during the life of the facility;

- The adequacy of SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents;
- Verification that the loss-of-coolant (LOCA) evaluation methods used are at a minimum under active NRC staff review and any open items can be reasonably left for later consideration in the FSAR, and there is reasonable assurance the proposed facility can be constructed and operated without undue risk to public health and safety; and
- Identification and plan for SSCs which require additional research and development to confirm the adequacy of the design and to resolve any outstanding safety questions.

For the selected limiting events, NUREG-0800 SRP Chapter 15 provides guidance for one acceptable method of review of transients and accidents, and associated analytical methods. Note that while it could be acceptable to use a bounding analysis to support siting of the facility, such an approach is design-specific and will likely require alternatives to existing staff guidance and regulatory exemptions. Therefore, any use of a bounding analysis approach will be reviewed on a case-by-case basis.

It is recognized that the design of the facility at the CP stage is not complete and the values of system parameters and setpoints used in the analysis will be preliminary in nature and subject to change in the future. Consistent with 10 CFR 50.35, some technical and design information may be reasonably left for a later stage of licensing. However, the staff must have confidence that any missing information and open safety questions can be satisfactorily resolved before completion of construction of the facility. Examples of items that could be reasonably left for later include:

- Evaluation of assumed non-limiting transients and accidents;
- Finalization of evaluation methods under active staff review at the time of CP application submittal;
- Additional research and testing necessary to satisfy 10 CFR 50.34(a)(8) and 50.35(a)(3),
- Finalization of system parameters and setpoints; and
- Development of technical specifications.

Structures, Systems, and Components

A CP should identify the safety categorization and design classification of the proposed facility SSCs. For components within the scope of 10 CFR 50.55a, a CP should also identify the edition of Codes and Standards proposed for the design. The staff should review the following:

- The design of components and supports within the jurisdiction of ASME, Section III, Division 1 should meet the applicable provisions of 10 CFR 50.55a;
- The proposed alternatives to ASME Codes and Standards should be consistent with the requirements in 10 CFR 50.55a(z);
- If utilizing the categorization in 10 CFR 50.69, “Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors,” the proposed standards for the design and treatment of components should be clearly identified for all four risk categories;

- A commitment to the following:
    - RG 1.100 (Ref. 14) for the seismic qualification of mechanical and electrical equipment, which endorses with few exceptions and clarifications the Institute of Electrical and Electronics Engineers (IEEE) Standard 344 and the ASME QME standard for “Qualification of Active Mechanical Equipment Used in Nuclear Power Plants”;
    - RG 1.136 (Ref. 15) for the design and qualification of concrete containment, which includes ASME Section III, Division 2 and ACI-359;
    - ASME Section III, Division 2 for the design and qualification of the spent fuel pool liner;
    - RG 1.142 (Ref. 16) for the design and qualification of the safety-related concrete structures other than containment, which includes ACI-349;
    - The latest NRC-endorsed edition of the American Institute of Steel Construction (AISC)/American National Standards Institute (ANSI) N690 for safety-related steel structures;
- For the cold-formed support members of conduit and cable trays, American Iron and Steel Institute (AISI) standard, "Specification for Design of Cold-Formed Steel Structural Members" is acceptable. For the hot-rolled support members of conduit and cable trays, AISC/ANSI N690 is acceptable; and
- The general construction of ducts is typically covered in Sheet Metal and Air Conditioning Contractors National Association SMACNA standards (typically used for non-safety-related applications). Safety-related HVAC ductwork is typically qualified to ASME AG-1, “Code on Nuclear Air and Gas Treatment.” For HVAC cold-formed member supports, the AISI Standard, "Specification for Design of Cold-Formed Steel Structural Members" is acceptable. For the hot-rolled structural members of the HVAC supports, AISC/ANSI N690 is acceptable.

Protective Coatings Systems

For proposed designs where protective coatings are relevant, the SRP provides guidance on the evaluation of the protective coating systems (paints) used inside the containment that are evaluated as to suitability for design basis accident (DBA) conditions. In a CP application, the staff reviews the applicant’s commitment to using protective coating systems to meet the SRP acceptance criteria. The SRP acceptance criterion is that a coating system to be applied inside a containment is acceptable if it meets the regulatory positions of RG 1.54 (Ref. 12) and the standards of ASTM D5144 (Ref. 1) and ASTM D3911 (Ref. 2). In cases where a CP applicant proposes an alternative to the guidance in the current revision of RG 1.54, the staff should focus on the following areas:

- Any exceptions to the Service Level definitions in RG 1.54 Section B should be justified, including any exceptions to the provisions and guidance in the associated ASTM standards (RG 1.54 regulatory position C.2.7);
- If exceptions are proposed to the Service Level definitions in RG 1.54, any assumptions about the coating’s properties and its response to a design-basis loss-of-coolant

accident, such as the form of debris, should be justified by references and supported by the coating qualification testing;
- Coatings qualification using ASTM D3911 should meet the minimum acceptance criteria in RG 1.54 regulatory position C.2.2;
- The coatings in-service monitoring program should meet the conditions in RG 1.54 regulatory position C.4.2, or exceptions are justified; and
- Thermal conductivity testing under D5144 should meet the exceptions in RG 1.54 regulatory position C.5.2.

Instrumentation and Control

In its development of a design specific review standard (DSRS) guidance (Ref. 6) for the NuScale small modular reactor design, the NRC incorporated some of the lessons learned from its review of large LWR designs. The guidance emphasizes fundamental instrumentation and control (I&C) design principles of independence, redundancy, predictability and repeatability, and diversity and defense-in-depth (D3). The current SRP guidance is system-focused and does not take advantage of such a unifying framework. The DSRS guidance aims to address all the significant aspects of the I&C design in a unified manner through this framework to minimize the repetition of the requirements in a system-focused approach. The structure of the DSRS guidance reflects an integrated I&C design using digital technology, introduces the use of an integrated hazards analysis approach to the I&C reviews, consolidates the various methods discussed in the current SRP, and provides a consistent, comprehensive, and systematic way to address the potential hazards associated with the I&C systems in a unified framework. Lastly, the guidance encompasses all relevant branch technical positions discussed in the current SRP and clarifies the interface between the I&C area and other disciplines, such as equipment qualification, human factors engineering, quality assurance, and reactor systems.

In evaluating a CP application, the reviewer should focus on the following elements of the I&C design:

- An overall I&C architecture that demonstrates adherence to the fundamental I&C design principles,
- Plant safety functions allocated to each of the safety-related I&C systems,
- Proposed communications between safety-related and non-safety-related I&C systems,
- Regulations that the I&C design intends to comply with,
- Regulations that the applicant intends to take exemption from or deems not applicable to its design, and
- Topical reports incorporated by reference in the application.

Electrical System Design

[The staff plans to provide information for this section at a later time.]

Radioactive Waste Management

The SRP Chapter 11 does not detail specific review guidance for radioactive waste management in a CP application. The staff should approach this review consistent with the SRP and the requirements in 10 CFR 50.34a as it applies to a CP; Appendix I to 10 CFR Part 50; general design criteria (GDCs) 60, 61, 63 and 64 in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50; and in consideration of the information that provides reasonable assurance that the applicant will comply with the requirements in 10 CFR Part 20.

APPENDIX

A. Resolution of Public Comments
B. References
C. Advanced Reactor Construction Permit Guidance

APPENDIX A

Resolution of Public Comments

APPENDIX B

References

1. American Society for Testing and Materials, ASTM D5144, “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants.”

2. American Society for Testing and Materials, ASTM D3911, “Standard Test Method For Evaluating Coatings Used In Light-Water Nuclear Power Plants At Simulated Design Basis Accident (DBA) Conditions.”

3. Memorandum from Frederick Brown, “Expectations for New Reactor Reviews,” U.S. Nuclear Regulatory Commission, August 29, 2018 (ADAMS Accession No.: ML18240A410).

4. Memorandum from Ho Nieh, “Applying the Principles of Good Regulation as a Risk-Informed Regulator,” U.S. Nuclear Regulatory Commission, October 15, 2019 (ADAMS Accession No.: ML19260E683).

5. U.S. Nuclear Regulatory Commission, “Draft white paper - Preapplication Engagement to Optimize Application Reviews,” October 15, 2020 (ADAMS Accession No.: ML20281A761).

6. U.S. Nuclear Regulatory Commission, “Design-Specific Review Standard for NuScale Small Modular Reactor Design,” August 4, 2016, (ADAMS Accession No.: ML15355A295).

7. U.S. Nuclear Regulatory Commission, COL/ESP-ISG-4, “Interim Staff Guidance on the Definition of Construction and on Limited Work Authorizations,” February 9, 2009 (ADAMS Accession No.: ML082970279).

8. U.S. Nuclear Regulatory Commission, NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.”

9. U.S. Nuclear Regulatory Commission, NUREG-2189, “Safety Evaluation Report Related to SHINE Medical Technologies, Inc., Construction Permit Application for a Medical Radioisotope Production Facility,” Docket No. 50-608, August 2016 (ADAMS Accession No.: ML16229A140).

10. U.S. Nuclear Regulatory Commission, NUREG-2229, “Safety Evaluation Report Related to Northwest Medical Isotopes, LLC, Construction Permit Application for a Radioisotope Production Facility,” Docket No. 50-609, May 2020 (ADAMS Accession No: ML20153A468).

11. U.S. Nuclear Regulatory Commission, “Management Directive and Handbook 8.4: Management of Backfitting, Issue Finality, and Information Requests,” September 20, 2019 (ADAMS Accession No.: ML18093B087).

12. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.54, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” Revision 3, April 2017 (ADAMS Accession No.: ML17031A288).

13. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (LWR Edition),” Revision 3, 1978 (ADAMS Accession No.: ML011340122).

14. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.100, “Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants,” Revision 3, May 2020 (ADAMS Accession No.: ML19312C677).

15. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.136, “Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments,” Revision 3, March 2007 (ADAMS Accession No.: ML070310045).

16. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.142, “Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments),” Revision 3, May 2020 (ADAMS Accession No.: ML20141L613).

17. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.206, “Regulatory Guide for Combined License Applications for Nuclear Power Plants,” Initial Issuance, June 2007 (ADAMS Accession No.: ML070720184).

18. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.206, “Regulatory Guide for Combined License Applications for Nuclear Power Plants,” Revision 1, October 2018 (ADAMS Accession No.: ML18131A181).

Appendix C
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[2]) 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. NRC staff should continue to consult other established guidance documents, as applicable, to complete reviews of non-LWR applications.

[2] 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.

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

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 plant structures, systems, and components (SSCs) modelled in the probabilistic risk assessment (PRA) 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”, and *draft TICAP ISG-XXX, Section 3.1.X.*

v. A summary of the approach used in conducting the safety analysis, including Licensing Basis Events (LBEs) including Design Basis Accident (DBA), 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.

For applications using the LMP approach, the staff should refer to draft *TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

2. Generic Analysis

a. Source Terms

The 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 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), and (for applications using the LMP approach) *draft TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

b. PRA
The staff should review how the applicant’s PRA is or will be used to support the analysis in the application. The application should summarize the scope, methodology, and pedigree of the PRA, to include what SSCs and human actions are modeled, and the scope and capability category to which the PRA was completed for the purposes of a CP. The pedigree is intended to be (i) a statement of compliance (with any exceptions) with the non-LWR PRA standard, ASME/ANS RA-S-1.4-2021, and the draft NRC white paper *“Demonstrating the Acceptability of Probabilistic Risk Assessment Results Used to Support Advanced Non-Light Water Reactor Plant Licensing”* (ML21015A434), the manner in which the standard was used, and the findings of PRA peer review conducted in accordance with NEI 20-09, Revision 1, “Performance of PRA Peer Reviews Using the ASME/ANS Advanced Non-LWR PRA Standard” (ML20302A115), or (ii) an alternative means of demonstrating PRA technical acceptability. For applications using the LMP approach, the staff should refer to *draft TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

c. Safety and Accident Analysis Methodologies and Associated Validation
The staff should review the safety and accident analysis methodologies and associated validation used by the applicant. The staff should review the description of plans to perform safety and accident analyses that include testing of applicable SSCs and validation and verification of associated engineering computer programs. The analysis plans need to include development of associated methodologies and applications of those methods which include but are not limited to event specific analysis methodologies, scaling methodology, setpoint methodology, reactor coolant analysis methodology, core design methodology, and reactivity control methods. The analysis plans need to include a test plan and test program as well as equipment qualification methodology to ensure appropriate verification and validation of the engineering computer programs, consistent to meet 10 CFR 50.43(e) for the future submittal of an OL application and following the guidance in RG 1.203, “Transient and Accident Analysis Methods” (ML053500170). For applications using the LMP approach, the staff should refer to *draft TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

d. Site Information
The 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) *draft TICAP ISG-XXX, Section 3.1.X, 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 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 Design Basis Accidents (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 following:

a. Discussion of selected DBAs. The 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 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.

iii. Experimental data reduction and/or correlation development is or will be performed using standard statistical techniques.

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. The staff should review the modeling of changes in chemical form as the releases are processed by mitigating systems to the environment from the site during the entire period of the DBA as a function of time.

e. Discussion of the meteorological characteristics of the proposed site used in the accident analysis including the site-specific 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), low population zone (LPZ) and control room (if operators 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 both frequency and 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 staff should perform an independent confirmatory radiological consequence analysis using pertinent information in the application to determine whether the proposed site meets the radiological consequence evaluation factors identified in 10 CFR 50.34(a)(1).

For applications using the LMP approach, the staff should refer to draft *TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

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.

For applications using the LMP approach, the staff should refer to draft *TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

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.

For applications using the LMP approach, the staff should refer to draft *TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

5. Safety Functions, Design Criteria, and SSC Categorization

a. Principal Design Criteria

The 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.

For applications using the LMP approach, the staff should refer to draft *TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

b. Safety-Related (SR) SSCs
The 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.

For applications using the LMP approach, the staff should refer to draft *TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

c. Complementary Design Criteria
The 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.

For applications using the LMP approach, the staff should refer to draft *TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

d. Non-Safety-Related with Special Treatment (NSRST) SSCs
The 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.

For applications using the LMP approach, the staff should refer to draft *TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

e. SSC Categorization Process
The 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.”

For applications using the LMP approach, the staff should refer to draft *TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

6. Safety-Related SSC Criteria and Capabilities
Refer to NEI 18-04 for a definition of SR SSCs. The 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 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 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.

    For applications using the LMP approach, the staff should refer to draft TICAP ISG-XXX, Section 3.1.X for additional information regarding expected CP application content in this area.

7. Non-Safety Related with Special Treatment (NSRST) SSC Criteria and Capabilities
Refer to NEI 18-04 for a definition of NSRST SSCs. The 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.

For applications using the LMP approach, the staff should refer to *draft TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

8. Plant Programs
   The 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. For applications using the LMP approach, the staff should refer to *draft TICAP ISG-XXX, Section 3.1.X* for additional information regarding expected CP application content in this area.

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."*

10. Control of Occupational Dose
    For guidance regarding specific information content refer to *draft ARCAP ISG, "Control of Occupational Dose."*

11. Organization
    For guidance regarding specific information content refer to *draft ARCAP ISG, "Organization."*

12. Initial Startup Program
    For guidance regarding specific information content refer to *draft ARCAP ISG, "Initial Startup Program."*

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)[3], 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. [4]

16. Aircraft Impact

The staff should review the applicant's analysis of aircraft impact per 10 CFR 50.150 which 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 refer to 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

[3] 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.
[4] Ibid

for use in satisfying the NRC requirements in 10 CFR 50.150(a) regarding the assessment of aircraft impacts for new nuclear power reactors.

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 staff should review the CP application information on fuel qualification (FQ) to determine if a reasonable plan for FQ has been proposed, or in some cases completed. Accordingly, the application may contain final design information or preliminary design information. The guidance in this section assumes that the application contains preliminary design information along with a description of what is needed to finalize the design (the FQ plan) and commitments to complete the described work.
The objective of fuel qualification is to demonstrate that the fuel performance is as described in the facility safety analysis.

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).

To evaluate the applicant's FQ plan the application needs to describe the fuel design in sufficient detail to provide an understanding of:

- its form (U, Pu, U/Pu, oxide, nitride, carbide, metal, TRISO, etc.),
- its composition (enrichment, cladding material, TRISO pebbles, TRISO prismatic elements, TRISO particles in solution, etc.),
- its proposed operating envelope (power density, design burnup level, neutron spectra, fluence, transients, etc.),
- its proposed performance (acceptance criteria, radionuclide retention characteristics, etc.),
- its physical description (pellet dimensions and density, cladding dimensions, TRISO pellet description, TRISO particle description, etc.).

The reviewer should determine if fuel of the proposed type has previously been used and/or tested in-reactor or ex-reactor and, if so, the extent of the operating experience (burnup achieved, power density, steady state and transient performance, etc.) available and the fuel performance observed. Based on the above, the staff should determine the gaps in information needed to support a final design and if the application provides a reasonable plan for obtaining the needed information. In some cases, a version of the FQ plan may have been reviewed during pre-application interactions with the staff. If so, the reviewer should determine what the results of the pre-application review were and factor those into the current review.

To determine the adequacy of the applicant's FQ plan, the following should be considered:

- What are the fuel performance criteria assumed in the safety analysis?
- Has fuel of this design been used before? If so, where and under what operating parameters (power density, burnup level, transients, enrichment, neutron spectra, fluence, etc.)? How does this experience compare to the proposed fuel performance criteria? Where there are differences, has the applicant proposed to obtain additional data to fill in the gaps?
- What has been the experience with the fuel's performance (damage, failure and radionuclide retention during steady state and transient conditions)? What additional data is needed to cover the complete range of proposed operating conditions? Has the applicant proposed to obtain this data?
- Has the fuel shown changes in properties (thermal conductivity, geometry, cladding performance) as a result of irradiation? Is additional data needed to cover the operating conditions of the proposed fuel design? Has the applicant proposed to obtain this data?
- Has fuel of this design been previously used with the coolant proposed for the reactor design? If so, how do the conditions (coolant temperatures and flowrate, impurities in the coolant, chemical reactions, corrosion, erosion, irradiation effects on

the coolant, etc.) associated with the operating experience compare to the planned operating envelope? Has the fuel been shown to be compatible with the coolant or are there effects that could impact safety? Where there is missing data, has the applicant proposed to obtain additional data to fill in the gaps?

- Does the applicant have fuel performance analytical tools? Do they cover the range of operating conditions (steady state and transient) proposed for the fuel? What validation has been done to demonstrate that the analytical tools predict the fuel performance? Has the applicant proposed to obtain additional data to cover any gaps?

The applicant's proposed FQ plan for developing additional data to fill in the gaps should be reviewed for completeness and technical adequacy. This should include:

- Obtaining sufficient data to demonstrate that the fuel performance is satisfactory for the power density, burnup, transients and coolant associated with the reactor design. Of particular interest should be the conditions under which existing data was obtained and additional data will be obtained. Where the data will be obtained and how it will be obtained should also be described. For fuel irradiation in a test reactor or another power reactor, how are any differences in test conditions between the proposed reactor design and the test conditions to be accounted for (e.g., does the test data account for the effects of higher energy neutrons or fluence that may be associated with the proposed design)? Are the planned power densities and burnup sufficient to cover the range of conditions expected? Do the transient tests cover the range of conditions included in the design? Will the test program, in conjunction with analysis, be sufficient to test the proposed criteria on fuel performance?
- Will analysis be used to extrapolate test data to conditions not tested? Has sufficient justification for the extrapolation been provided or included in the FQ plan?
- What QA program will be applied to the test program to ensure accurate data is obtained?
- What acceptance criteria are to be applied in evaluating the adequacy of the test data (steady state operation and transient performance)?
- What is the plan and schedule for completing the work necessary to fill in the gaps? Will the plan be completed on a schedule that allows the staff sufficient time to review the final design?

It may be the case that some or all of the FQ plan is described in a separate document, in which case the application should provide a reference to the document and summarize the FQ plan in the application. In evaluating the FQ plan, the staff should also request the applicant to identify the safety classification of the fuel. Is fuel integrity an essential part of the safety case or is fuel failure assumed in the safety analysis? If fuel integrity is critical to the safety case, the reviewer should determine how the applicant intends to ensure the quality of the fuel will be consistent with its design. The fuel procurement specification should describe the characteristics the fuel must have to be acceptable. The reviewer should determine how the applicant intends to ensure the as fabricated fuel complies with the procurement specification. This can include oversight of the fuel fabrication process, ensuring an inspection program is implemented that is capable of detecting deviations from

the procurement specification and requiring a comprehensive QA Program be applied to the fuel fabrication process.

If fuel failure is assumed in the safety analysis, the reviewer should determine if the proposed test program is sufficient to confirm that the fuel damage, fuel failure and their timing are as assumed in the safety analysis.

Upon completion of the review, the reviewer needs to conclude that the application provides a reasonable basis for the proposed FQ plan and, assuming that the applicant successfully completes the activities described in the plan, there is a reasonable expectation that the fuel can be qualified.

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.