DNRL-ISG-2022-XX

Safety Review of Light-Water Power-Reactor Construction Permit Applications

Draft Interim Staff Guidance

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DRAFT INTERIM STAFF GUIDANCE

SAFETY REVIEW OF LIGHT-WATER POWER-REACTOR CONSTRUCTION PERMIT APPLICATIONS

DNRL-ISG-2022-XX

PURPOSE

The U.S. Nuclear Regulatory Commission (NRC or Commission) staff is providing this interim staff guidance (ISG) to facilitate the safety review of light-water power-reactor construction permit (CP) applications.

BACKGROUND

The NRC anticipates the submission of power reactor CP applications within the next few years based on preapplication engagement initiated by several prospective applicants. The review of these applications falls within the two-step licensing process under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities," (Reference 1) and involves the issuance of a CP before an operating license (OL). The NRC last issued a power reactor CP in the 1970s. Most recently, the NRC issued combined construction and operating licenses (combined licenses or COLs) for power reactors through the one-step licensing process under 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants," (Reference 2) using the guidance in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: [Light-Water Reactor (LWR)] Edition" (SRP) (Reference 3); and Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," issued June 2007 (Reference 4). The NRC has periodically updated some of the SRP guidance and issued Revision 1 to RG 1.206, "Applications for Nuclear Power Plants," in October 2018 (Reference 5).

The licensing process under 10 CFR Part 50 allows an applicant to begin construction with preliminary design information instead of the final design required for a COL under 10 CFR Part 52. Although the two-step licensing process provides flexibility and allows a more limited safety review before construction, the design has less finality before the applicant commits to construction of the facility. The final safety analysis report (FSAR) submitted with the OL application should describe in detail the final design of the facility as constructed; identify the changes from the criteria, design, and bases in the CP preliminary safety analysis report (PSAR); and discuss the bases for, and safety significance of, the changes from the PSAR. Before issuing an OL, the NRC staff will review the applicant's final design in the FSAR to determine whether all the Commission's safety requirements have been met.

The SRP contains the NRC staff review guidance for LWR applications submitted under 10 CFR Part 50 or 10 CFR Part 52. In addition to the CP review guidance in the SRP, RG 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Revision 3, issued November 1978 (Reference 6), offers some insights on the level of detail that is required for the PSAR in support of the CP application, but these insights may be limited to the degree that the guidance does not account for subsequent requirements, NRC technical positions, or advances in technical knowledge. RG 1.206 provides guidance for 10 CFR Part 52 applications, including for early site permits (ESPs) and COLs, and includes insights on the level of detail needed for final design information if the CP applicant chooses to provide such information.

The NRC recently issued CPs for two nonpower production and utilization facilities (NPUFs)—SHINE Medical Technologies, Inc. (Reference 7), and Northwest Medical Isotopes, LLC (Reference 8). Some of the lessons learned from these reviews are applicable to the review of power reactor CP applications, as discussed below.

RATIONALE

This ISG focuses on the safety review of power reactor CP applications for any LWR design, including designs similar to those reviewed recently under 10 CFR Part 52, and may refer to the applicable guidance for the review of non-LWR designs. The NRC staff last reviewed a CP application for a power reactor about 40 years ago. Although the LWR CP application guidance in RG 1.70 dates from the 1970s and the staff developed the more recent LWR application guidance in RG 1.206 for 10 CFR Part 52 applications, these documents provide some insights on the level of detail to support an LWR CP application review as discussed above.

APPLICABILITY

All applicants for a CP for a light-water power-reactor under 10 CFR Part 50.

GUIDANCE

This ISG discusses some of the regulatory requirements for a CP, applicable review guidance in the SRP, and special topics related to CP applications. The appendix to this ISG supplements the SRP by clarifying the review of certain information in a CP application.

<u>Requirements for a Power Reactor Construction Permit Application</u>

A number of regulations apply to a power reactor CP application, including but not limited to the following:

- 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 10 CFR 50.34(a) on the PSAR
- 10 CFR 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors”

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

- 10 CFR 50.35, “Issuance of construction permits”
- 10 CFR 50.40, “Common standards”
- 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors”
- 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 50.150, “Aircraft impact assessment”
- 10 CFR Part 20, “Standards for protection against radiation”
- 10 CFR Part 100, “Reactor site criteria”

The following discussion focuses on information in the PSAR, certain regulations applicable to a CP, and the criteria for issuing a CP.

The regulations in 10 CFR 50.34(a) specify the minimum technical information in the PSAR accompanying a CP application, including preliminary design information and a description and safety assessment of the site on which the facility is to be located. As required by 10 CFR 50.34(a)(3), the preliminary design information must include the principal design criteria, the design bases and the relation of the design bases to the principal design criteria, and sufficient information on the materials of construction, general arrangement, and approximate dimensions for the staff to conclude that the final design will conform to the design bases with adequate margin for safety. In accordance with 10 CFR 50.34(a)(1), the site safety assessment must 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 (Reference 9).

The regulations in 10 CFR 50.34a require for a description of the preliminary design of equipment to maintain control of radioactive material in effluents produced during normal reactor operations, and the design objectives and means for keeping the levels of radioactive material in effluents as low as is reasonably achievable. Also, 10 CFR 50.34a requires a CP application to estimate the kinds and quantities of the principal liquid and gaseous radionuclides that would be released to unrestricted areas during normal reactor operations, as well as to describe the provisions for packaging and storing radioactive solid waste materials and shipping them offsite.

The regulations in 10 CFR 50.150 require CP applicants to perform a realistic design-specific assessment of the effects on the facility of the impact of a large, commercial aircraft and to identify and incorporate into the design those design features and functional capabilities to show that, with reduced operator actions, the criteria in 10 CFR 50.150(a)(1)(i)-(ii) are satisfied. SRP Section 19.5, “Adequacy of Design Features and Functional Capabilities Identified and Described for Withstanding Aircraft Impacts,” provides guidance acceptable to the staff for performing the licensing review. The NRC’s decision on an application subject to

10 CFR 50.150 will be separate from any NRC determination that may be made with respect to the adequacy of the impact assessment, which is not submitted to the NRC.[2]

If the application does not contain sufficient information for the NRC staff to approve all proposed design features, the CP may be issued if the NRC makes the findings listed in 10 CFR 50.35(a). The flexibilities in this regulation do not obviate the other requirements applicable to a CP, such as those in 10 CFR 50.34(a). The CP application will need to include sufficient information for the staff to conduct its review and evaluate the information against the applicable regulations.

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 were instead merged into an OL. 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" CP, but the criteria in 10 CFR 50.35(a) for issuing a CP remained the same as the criteria used to issue the former "provisional" CP (35 FR 5317; March 31, 1970). Historically, when issuing a power reactor CP under 10 CFR 50.35(a), the Commission authorized the construction of the facility described in the application and hearing record, in accordance with the principal architectural and engineering criteria and the commitments identified therein.[3]

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,

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2 "Consideration of Aircraft Impacts for New Nuclear Power Reactors," Volume 74 of the *Federal Register* (FR), page 28120 (74 FR 28120; June 12, 2009).

3 An example is the CP issued for the Shearon Harris Nuclear Power Plant (Reference 10). CPs also included permit conditions on specified issues.

> the proposed facility can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

If a novel design has not sufficiently progressed and certain information is not available at the time of CP application submission, the PSAR should provide the criteria and bases that will be used to develop the required information, the concepts and alternatives under consideration, and the schedule for completing the design and submitting the missing information. In general, the PSAR should describe the preliminary design of the facility in sufficient detail to enable the NRC staff to evaluate whether the facility can be constructed and operated without undue risk to the public health and safety. The NRC expects the CP application and PSAR to address all regulatory requirements applicable to a CP.

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, an applicant may provide more complete technical information in its CP application and thereby reduce the regulatory review in the subsequent OL review 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, including new or novel design features or unique specifications.[4] This request for approval would need more than preliminary information to support the NRC staff's review to approve such design features or specifications. In such a case, the NRC expects that the level of design information available in the application to support the approval of a proposed design feature would be the same level of design information available for a 10 CFR Part 52 COL application. RG 1.206 contains guidance on the level of design information that the NRC expects is available to support a COL application. 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 in 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 such approval and the approval is incorporated into the permit.

As described in 10 CFR 50.35(c), the NRC will not issue a license authorizing operation of the facility until (1) the applicant submits, as part of an OL application, its FSAR and (2) the Commission finds that the final design provides reasonable assurance that operation of the facility will not endanger public health and safety. 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. Before issuing an OL, the NRC staff will review the applicant's final design in the FSAR to determine whether it has met all the Commission's safety requirements. Based on this determination, the Commission would issue an OL, and the applicant may then

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4 The special topics section of this ISG discusses preapplication activities that have proven effective and essential in gaining an 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 application review.

operate the facility in accordance with the terms of the OL and the Commission's regulations under continued oversight by the NRC staff. Commission procedures include an opportunity for public hearings before the authorization of either facility construction or operation, and a mandatory hearing before issuance of a CP.

Light-Water-Reactor Safety Review Guidance

The SRP provides guidance to assure quality and predictability in the NRC staff's safety review of various licensing actions, including an LWR CP application. The SRP and the additional guidance included later in this document provide the NRC staff with an acceptable approach for verifying that the applicable requirements in the regulations for LWR applications are met. Implementation of the acceptance criteria contained in the SRP and the additional guidance in this document give assurance that an LWR design will comply with the Commission's regulations and provide adequate protection of public health and safety.

The NRC staff should review the risk-significant and safety-significant aspects of the application commensurate with their significance. The review should focus on those aspects of the design that contribute most to safety and minimize attention on issues of low risk or safety significance. An aspect of a design can be more or less risk-significant than that design aspect typically is for other reactors, thereby justifying more or less scrutiny than is typical for that design aspect.

Consistent with the NRC's use of risk-informed decision-making, the NRC staff should integrate risk insights with traditional engineering approaches to provide better reasoned regulatory decisions and appropriately disposition issues that arise in all regulatory matters, including licensing activities. This approach also implements the direction in the Commission's Probabilistic Risk Assessment (PRA) Policy Statement (60 FR 42622; August 16, 1995), which states, in part, "The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy." SRM-SECY-98-144, "White Paper on Risk-Informed and Performance-Based Regulation" (Reference 11), discusses the terms and concepts involved in the PRA policy statement and how these concepts are to be applied to NRC rulemaking, licensing, inspection, assessment, enforcement, and other decision-making.

Applications for licenses under 10 CFR Part 50 and 10 CFR Part 52 typically follow the structure of the SRP to efficiently support the NRC 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 NRC staff will use the methods described in the SRP and this document to evaluate the application's conformance with the Commission's regulations. If an applicant proposes to use an alternative approach or standard in its application, the NRC staff will evaluate the alternative approach or standard to ensure that it demonstrates compliance with the Commission's regulations. In many cases, the deterministic guidance in the SRP represents one of multiple acceptable ways to meet higher level regulatory requirements, such as those presented in the general design criteria in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50. In these cases, the NRC staff has the latitude to determine appropriate criteria for evaluating the alternative approach or standard for acceptability to ensure that the alternative approach or standard provides reasonable assurance of adequate protection of public health and safety, and that the bases for, and limitations on, the NRC staff's approval are clearly delineated.

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 appendix to this ISG provides clarifying CP review guidance for those SRP sections that combined CP and OL review guidance or where more information on the approach for reviewing preliminary design information is needed.

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 Part 50 and 10 CFR Part 52. Although the guidance in RG 1.70 dates from the 1970s and the guidance in RG 1.206 applies to 10 CFR Part 52 applications, the information in these RGs supports a CP application structure consistent with the SRP, helps to ensure the completeness of information in the application, and provides insights on what information in the application would support the NRC staff's safety review and evaluation.

The initial issuance of RG 1.206 (Reference 4) provides guidance on the format, content, and level of detail for a COL referencing a final design in a layout similar to RG 1.70. Revision 1 to RG 1.206 (Reference 5) expands the scope of the guidance to include applications for design certifications (DCs), ESPs, and limited work authorizations (LWAs) and removes the description of the technical information included in the safety analysis report that is contained in the SRP.

Although the RGs provide insights, the NRC staff should use the SRP to guide its review as superseded or supplemented by new or revised regulations, other regulatory guidance, NRC staff analyses of previous applications, and other published NRC staff positions, being mindful of the Commission policy in Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," dated September 20, 2019 (Reference 12), on using, in appropriate circumstances, the same reasoned decision-making process as employed for forward fits. In addition, the NRC staff should approach its review consistent with the expectations for new reactor reviews documented in the memorandum from Frederick Brown dated August 20, 2018 (Reference 13) and apply the principles of good regulation discussed in the memorandum from Ho Nieh dated October 15, 2019 (Reference 14).

Special Topics

This section discusses the relationship between the CP and OL reviews; the purposes and benefits of preapplication activities; the lessons learned from recently issued nonpower reactor CPs; the approach for reviewing concurrent license applications and applications incorporating prior NRC approvals; the potential effect of ongoing regulatory activities on CP reviews; and the licensing requirements for byproduct, source, or special nuclear material.

Relationship between the Construction Permit and Operating License Reviews

The approach to reviewing a CP application is intended to differ from the more recent COL application reviews in which an applicant provides all technical information on the final facility design to support the Commission findings for issuance of a COL under 10 CFR Part 52. As discussed in the original proposed 10 CFR 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 NRC staff's review and evaluation of the proposed design of a facility provided in a CP application constitutes 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. Consistent with recently issued CPs, CP conditions of a confirmatory nature focus on the additional information needed to address certain matters related to the safety of a final design and require the applicant to submit periodic reports on such information to the NRC before the completion of construction.

Purposes and Benefits of Preapplication Activities

Preapplication activities have proven effective and essential for gaining an early understanding of the applicant's plans and its proposed facility design, supporting the early resolution of unique design aspects of the facility, and preparing resources for the application review. These interactions were key for the recently issued permits for the construction of medical radioisotope facilities as NPUFs licensed under 10 CFR Part 50. Insights gained from such interactions may bridge gaps in the existing SRP review guidance for a particular facility design.

The staff has developed a draft white paper to provide information to advanced reactor developers on the benefits of robust preapplication engagement in order to optimize application reviews. The staff is in the process of capturing this white paper in advanced reactor content of application project (ARCAP) guidance. Although directed to the advanced reactor community, the preapplication engagement guidance, when issued as final as part of the ARCAP guidance development process, may be applicable to LWR license applicants and, if fully executed, will enable the NRC staff to offer more predictable and shorter schedules and other benefits during the review of a reactor license application.

Consistent with regulatory requirements and Commission policy statements, the NRC staff should more fully integrate the use of risk insights into preapplication activities by aligning its review focus and resources to the risk-significant SSCs and other aspects of the design that contribute most to safety and thereby enhance the efficiency of the review process.

Lessons Learned from Recently Issued Construction Permits

As noted above, the NRC has issued CPs for two NPUFs licensed under 10 CFR Part 50: (1) SHINE Medical Technologies, LLC, in February 2016; and (2) Northwest Medical Isotopes, LLC, in May 2018. The NPUF lessons learned described below 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 and other regulations applicable to a CP. However, those drawing lessons from recent NPUF reviews should consider the different technologies involved and the much more limited set of safety requirements that apply to an NPUF as opposed to a power reactor.

Lessons learned from the review of these NPUF CP applications include the following:

- Preapplication engagement is key to providing near-term guidance to the applicant.
- Early interactions support a common understanding of the information needed in the PSAR and the information that could reasonably be left for the FSAR accompanying the OL application, such as operational program descriptions.
- If the PSAR includes preliminary or limited descriptions of the facility's programs or SSCs, the NRC staff may accept and approve the application with regulatory commitments from the applicant to provide complete information in its OL application.
- The NRC staff's CP safety review is focused on ensuring the appropriate use of analysis methodologies to meet the requirements in the regulations.

In the safety evaluations related to the CPs issued, the NRC staff noted the applicant's regulatory commitments for the resolution of items that were not necessary for the issuance of a CP, but that the applicant should address in the FSAR submitted with an OL application. The CPs included conditions to ensure that the permit holder informed the NRC of safety-significant areas of construction before 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 the final design and required the applicant to submit to the NRC, before the completion of construction, periodic reports on such information.

The NRC staff should consider the lessons learned in its approach to the review of a reactor CP application and be mindful of the different regulations applicable to a power reactor and the existing NRC staff review guidance in the SRP as supplemented by this ISG.

Concurrent Applications

A CP application may be accompanied by an application for an LWA. For the LWA review, the NRC staff should refer to the guidance in RG 1.206, Revision 1 (Reference 5), related to the definition of construction and LWAs.

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

The NRC staff notes the inherent complications associated with concurrent CP and OL reviews. For example, as a result of the OL review, a need to reclassify SSCs (i.e., from not 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, addressing this reclassification could require extensive rework of both the CP and OL applications.

Construction Permit Application Incorporating Prior NRC Approvals

A CP application may incorporate prior NRC approvals by reference, including a standard design approval (SDA), a DC, or an ESP. Each of these approvals is supported by an NRC 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 10 CFR Part 52 approval.

If the NRC 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 NRC staff's CP review with regard to the referenced material would generally be limited to an evaluation of (1) how the CP application addresses the referenced approval conditions and additional requirements and restrictions, and (2) any deviations from the referenced material that are subject to prior NRC review. The NRC staff's CP review will focus on the portions of the application not receiving prior NRC approval.

For a CP application referencing an ESP, the NRC staff's review and evaluation may be more extensive in that the NRC 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 update 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 a DC, the NRC staff's review and evaluation may focus 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 an essentially complete design, while an SDA may approve only major features of the design. This difference may affect the level of design information that the CP application might need to include. Also, Section IV.B in all issued DC 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." The basis for this restriction is discussed in the final rule for the U.S. Advanced Boiling Water Reactor DC.[5]

For a CP application referencing an ESP and an SDA or a DC, the NRC staff's review and evaluation would generally focus 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 Part 50 and 10 CFR Part 52 through rulemaking consistent with Commission direction described in SRM-SECY-15-0002, "Proposed Updates of Licensing Policies, Rules, and Guidance for Future

[5] "Standard Design Certification for the U.S. Advanced Boiling Water Reactor Design," 62 FR 25800 (May 12, 1997).

New Reactor Applications" (Reference 15). 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). The NRC staff should continue to monitor the progress of the 10 CFR Part 50 and 10 CFR Part 52 rulemaking since a CP applicant must comply with the applicable regulations that are in effect at the time the NRC issues the CP.

In the SRM-SECY-15-0002, the Commission also confirmed that the Commission's "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" (50 FR 32138; August 8, 1985) (Severe Accident Policy Statement) and other Commission direction identified in SECY-15-0002 (Reference 16) apply to new 10 CFR Part 50 power reactor applications in a manner consistent with 10 CFR Part 52 design and license applications. Consistent with the Commission's Severe Accident Policy Statement, an applicant submitting a new design for NRC approval could address severe accidents acceptably if it:

- Demonstrated compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island (TMI) requirements described in 10 CFR 50.34(f);
- Demonstrated technical resolution of all applicable Unresolved Safety Issues and the medium- and high-priority Generic Safety Issues, including a special focus on assuring the reliability of decay heat removal systems and the reliability of electrical supply systems;
- Completed a Probabilistic Risk Assessment (PRA) and consideration of the severe accident vulnerabilities the PRA exposes along with the insights that it may add to the assurance of no undue risk to public health and safety; and
- Completed a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by PRA.

A discussion of the staff's proposal to apply this policy statement and other Commission direction to new 10 CFR Part 50 power reactor applications is provided in SECY-15-0002. The other Commission direction discussed in SECY-15-0002 includes Commission direction in response to SECY-89-013, SECY-90-016, and SECY-93-087.

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 COL, a CP, an OL, a DC, an SDA, or a manufacturing license. Many of the topics covered in the ARCAP guidance are also applicable to LWR designs, including updated siting guidance. The NRC staff may consider the ARCAP guidance, when final, for applicability to the review of an LWR CP application.

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

This ISG does not provide review guidance on the licensing requirements for byproduct, source, or special nuclear material under 10 CFR Part 30, "Rules of general applicability to domestic licensing of byproduct material"; 10 CFR Part 40, "Domestic licensing of source material"; and 10 CFR Part 70, "Domestic licensing of special nuclear material." The CP applicant may address the applicable materials licensing requirements with its CP application (in accordance with 10 CFR 50.31, "Combining applications") or separately from the CP application.

IMPLEMENTATION

The NRC staff will use the information discussed in this ISG to supplement the guidance in the SRP to determine whether regulations applicable to a CP are met, including the requirements in 10 CFR 50.35 for the issuance of a CP.

BACKFITTING AND ISSUE FINALITY DISCUSSION

Discussion to be provided in final ISG.

CONGRESSIONAL REVIEW ACT

Discussion to be provided in final ISG.

FINAL RESOLUTION

The staff will transition the information and guidance in this ISG into the SRP, as appropriate, when the staff completes the next periodic update of applicable SRP sections. Following the transition of all pertinent information and guidance in this document into the SRP, this ISG will be closed.

REFERENCES

1. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities."

2. 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants."

3. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition." (https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr0800/index.html)

4. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.206, "Regulatory Guide for Combined License Applications for Nuclear Power Plants," Initial Issuance, June 2007 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML070720184).

5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.206, "Applications for Nuclear Power Plants," Revision 1, October 2018 (ADAMS Accession No. ML18131A181).

6. 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, November 1978 (ADAMS Accession No. ML011340122).

7. U.S. Nuclear Regulatory Commission, "SHINE Medical Technologies, Inc., Docket No. 50-608, Medical Isotope Production Facility Construction Permit," Construction Permit No. CPMIF-001, February 29, 2016 (ADAMS Accession No. ML16041A471).

8. U.S. Nuclear Regulatory Commission, "Northwest Medical Isotopes, LLC, Docket No. 50-609, Medical Radioisotope Production Facility Construction Permit," Construction Permit No. CPMIF-002, May 9, 2018 (ADAMS Accession No. ML18037A468).

9. 10 CFR Part 100, "Reactor site criteria."

10. Correspondence from Roger S. Boyd, "Issuance of Construction Permits - Shearon Harris Nuclear Power Plant, Units 1, 2, 3 and 4," U.S. Nuclear Regulatory Commission, January 27, 1978 (ADAMS Accession No. ML020560123).

11. U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum (SRM), SRM-SECY-98-144, "White Paper on Risk-Informed and Performance-Based Regulation," March 1, 1999 (ADAMS Accession No. ML003753601).

12. U.S. Nuclear Regulatory Commission, Management Directive and Handbook 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," September 20, 2019 (ADAMS Accession No. ML18093B087).

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

14. 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).

15. U.S. Nuclear Regulatory Commission, Staff Requirements Memorandum, SRM-SECY-15-0002, "Staff Requirements – SECY-15-0002 – Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications," September 22, 2015 (ADAMS Accession No. ML15266A023).

16. U.S. Nuclear Regulatory Commission, SECY-15-0002, "Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications," January 8, 2015 (ADAMS Accession No. ML13277A420).

APPENDIX

CLARIFICATIONS TO THE EXISTING REVIEW GUIDANCE IN NUREG–0800

An applicant may use the information in Regulatory Guide (RG) 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” (Reference 1), RG 1.206, “Regulatory Guide for Combined License Applications for Nuclear Power Plants,” issued June 2007 (Reference 2), and RG 1.206, Revision 1, “Applications for Nuclear Power Plants,” issued October 2018 (Reference 3), on the format, content, and level of detail to develop a license application submitted under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, “Domestic licensing of production and utilization facilities,” (Reference 4) or 10 CFR Part 52, “Licenses, certifications, and approvals for nuclear power plants” (Reference 5). Although the guidance in RG 1.70 dates from the 1970s and the guidance in RG 1.206 is relevant to license applications submitted under 10 CFR Part 52, the information in these RGs supports a CP application structure consistent with NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP) (Reference 6); helps to ensure the completeness of the information in the application; and provides insights on the information that the application needs to support the U.S. Nuclear Regulatory Commission (NRC) staff’s safety review and evaluation.

The NRC staff should be familiar with these RGs, approach the CP application consistent with the guidance in the SRP and this ISG, and be aware that 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 NRC staff should guide its review using the SRP as superseded or supplemented by new or revised regulations, other regulatory guidance, NRC staff analyses of previous applications, and other published NRC staff positions, being mindful of the Commission policy in Management Directive 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests,” dated September 20, 2019 (Reference 7), on using, in appropriate circumstances, the same reasoned decision-making process as employed for forward fits. In addition, the NRC staff should approach its review consistent with the expectations for new reactor reviews documented in the memorandum from Frederick Brown dated August 20, 2018 (Reference 8) and apply the principles of good regulation discussed in the memorandum from Ho Nieh dated October 15, 2019 (Reference 9).

The NRC staff should review risk-significant and safety-significant aspects of the application commensurate with their significance. The NRC staff’s review should focus on those aspects of the design that contribute most to safety and minimize attention on issues of low risk and safety significance. An aspect of a design can be more or less risk-significant than that design aspect typically is for other reactors, thereby justifying more or less scrutiny than is typical for that design aspect.

Consistent with the NRC’s use of risk-informed decision-making, the NRC staff should integrate risk insights with traditional engineering approaches in providing better reasoned regulatory decisions to appropriately disposition issues that arise in all regulatory matters, including licensing activities. This approach also implements the direction in the Commission’s

Probabilistic Risk Assessment (PRA) Policy Statement (Volume 60 of the *Federal Register* (FR), page 42622 (60 FR 42622); August 16, 1995), which states, in part, “The use of PRA technology should be increased in all regulatory matters to the extent supported by the state of the art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.” SRM-SECY-98-144, “White Paper on Risk-Informed and Performance-Based Regulation” (Reference 10), discusses the terms and concepts involved in the PRA policy statement and how these concepts are to be applied to NRC rulemaking, licensing, inspection, assessment, enforcement, and other decision-making.

This appendix provides clarifying and supplemental guidance to the SRP for CP reviews applicable to those SRP sections that combined CP and operating license (OL) review guidance or where more information on the approach for reviewing preliminary design information is needed.

Finally, the NRC staff should note that the information in this appendix is not intended to include all topics expected and reviewed in a CP application.

Siting

The NRC 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 human-related hazards. The NRC staff’s review of these topics should determine how these site characteristics have influenced plant design and operating criteria and should examine the adequacy of the site characteristics from a safety viewpoint. SRP Chapter 2, “Site Characteristics and Site Parameters,” provides guidance for reviewing these technical areas. SRP Chapter 13 includes guidance on the requirements of 10 CFR Part 100, “Reactor site criteria,” (Reference 11) related to the development of the security and emergency plans. The NRC expects that the applicant will completely characterize the site selected for construction. Also, the application should include a commitment that, if an unexpected feature 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 the radiological consequence analyses in a CP application with preliminary design information, the NRC staff should use the guidance in SRP Section 15.0.3, “Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors,” and consider the applicant’s use of bounding assumptions to account for uncertainty in the final design and the potential for different methods presented in the final safety analysis report accompanying the OL application. The NRC staff should approach the review of safety and siting analyses commensurate with the specificity of the design details and safety assessment in the application, focusing on the major safety features and components in the design that support site suitability. In a CP review for a preliminary design, the NRC 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. Consistent with 10 CFR 50.35, “Issuance of construction permits,” some technical and design information may reasonably be left for a later stage of licensing. However, the NRC staff must have confidence that any missing information and open safety questions can be resolved satisfactorily before the completion of facility construction.

Transient and Accident Analyses

Consistent with the guidance in SRP Chapter 15, “Transient and Accident Analysis,” 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 contribute significantly 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 CP reviews of facilities to support a finding that the proposed facility can be constructed and operated without undue risk to public health and safety as required by 10 CFR 50.34, “Contents of applications; technical information,” and 10 CFR 50.35.

It is essential that all credible design-basis transients and 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, the application does not document analytical methods but instead refers to a vendor topical report. Examples of such methods for light-water-reactor (LWR) designs include departure from nucleate boiling correlation development, subchannel analysis, system transient analysis, analysis of reactivity-initiated accidents, and loss-of-coolant accident (LOCA) analysis. If applicants use techniques previously considered and approved by the NRC, the NRC staff verifies the previously approved method is applicable and stipulated limitations and conditions are satisfied. However, if new methods are involved, the staff reviews topical reports and other information that describe the method of analysis. 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 NRC staff 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 NRC staff verifies that the applicant systematically analyzed and evaluated the limiting events in each category using a detailed quantitative analysis. At a minimum, the NRC staff should ensure the preliminary safety analysis report includes all the information required by 10 CFR 50.34, with a focus on the following:

- evaluations of the design and structure, system, and component (SSC) performance resulting from facility operation

- 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 accident consequences

- verification that the LOCA evaluation methods used are approved and applicable to the design

- verification that non-LOCA evaluation methods are at a minimum under active NRC staff review and any open items can reasonably be left for later consideration in the final safety analysis report, and there is reasonable assurance the proposed facility can be constructed and operated without undue risk to public health and safety

- identification and plan for SSCs that require additional research and development to confirm the adequacy of the design and to resolve any outstanding safety questions

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

The NRC recognizes that the facility's design 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 reasonably be left for a later stage of licensing. However, the NRC staff must have confidence that any missing information and open safety questions can be resolved satisfactorily before completion of construction of the facility. Examples of items that could reasonably be left for later include the following:

- finalization of analyses for regulated beyond-design-basis events (e.g., station blackout, anticipated transients without scram)

- finalization of evaluation methods under active NRC staff review at the time of CP issuance

- additional research and testing necessary to satisfy 10 CFR 50.34(a)(8) and 10 CFR 50.35(a)(3)

- finalization of system parameters and setpoints

- 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, "Codes and standards," a CP should also identify the edition of codes and standards proposed for the design. Consistent with the guidance in SRP Chapter 3, "Design of Structures, Components, Equipment, and Systems," the NRC staff should review the following:

- The design of components and supports within the jurisdiction of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), 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 using 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.

- The applicant should commit to the following or justify an alternative:

  - RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," Revision 3, issued May 2020 (Reference 12), for the seismic qualification of mechanical and electrical equipment, which endorses with few exceptions and clarifications, Institute of Electrical and Electronics Engineers (IEEE) Standard 344, "IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations," (Reference 13) and ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities" (Reference 14)

  - RG 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments," Revision 3, issued March 2007 (Reference 15), for the design and qualification of concrete containment, which includes ASME Code, Section III, Division 2, and American Concrete Institute (ACI)-359, "Concrete Containments for Nuclear Reactors" (Reference 16)

  - ASME Code, Section III, Division 2, for the design and qualification of the spent fuel pool liner

  - RG 1.142, "Safety Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," Revision 3, issued May 2020 (Reference 17), for the design and qualification of the safety-related concrete structures other than containment, which includes ACI-349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary" (Reference 18)

  - The latest NRC-endorsed edition of the American Institute of Steel Construction (AISC)/American National Standards Institute (ANSI) N690, "Specification for Safety-Related Steel Structures for Nuclear Facilities" (Reference 19)

- For the cold-formed support members of conduit and cable trays, American Iron and Steel Institute (AISI) S100-16, "North American Specification for the Design of Cold-Formed Steel Structural Members," (Reference 20) is acceptable. For the hot-rolled support members of conduit and cable trays, AISC/ANSI N690 is acceptable.

- The general construction of ducts is typically covered in Sheet Metal and Air Conditioning Contractors National Association standards (typically used for non-safety-related applications). Safety-related heating, ventilation, and air conditioning

(HVAC) ductwork is typically qualified to ASME AG-1, “Code on Nuclear Air and Gas Treatment” (Reference 21). For HVAC cold-formed member supports, AISI S100-16 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, SRP Section 6.1.2, “Protective Coating Systems (Paints)—Organic Materials,” provides guidance on evaluating the protective coating systems (paints) used inside the containment that are evaluated as to suitability for design-basis accident conditions. In a CP application, the NRC staff reviews the applicant’s commitment to using protective coating systems to meet the SRP acceptance criterion. 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, “Service Level I, II, III, and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants,” Revision 3, issued April 2017 (Reference 22), and the standards of American Society for Testing and Materials (ASTM) D5144, “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants” (Reference 23), and ASTM D3911, “Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions” (Reference 24). If a CP applicant proposes an alternative to the guidance in the current revision of RG 1.54, the NRC 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 the applicant proposes exceptions to the service level definitions in RG 1.54, any assumptions about the coating’s properties and its response to a design-basis LOCA, 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 inservice monitoring program should meet the conditions in RG 1.54, Regulatory Position C.4.2, or exceptions should be justified.
- Thermal conductivity testing under ASTM D5144 should meet the exceptions in RG 1.54, Regulatory Position C.5.2.

Instrumentation and Control

In its development of design-specific review standard (DSRS) guidance (Reference 25) 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. The guidance in SRP Chapter 7, “Instrumentation and Controls,” 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 SRP Chapter 7; 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 SRP Chapter 7 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 NRC staff 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 applicant intends to comply with for the I&C design
- regulations that the applicant intends to take exemption from or deems not applicable to its design
- topical reports incorporated by reference in the application

Electrical System Design

For proposed designs that rely on electrical power, SRP Chapter 8, “Electric Power,” provides detailed guidance on the evaluation of electrical power sources to support normal, abnormal, and accident conditions. At the CP stage, the NRC staff should review the classification of SSCs of the proposed design and the portions of the onsite and offsite power systems that are designated Class 1E and non-Class 1E and the justification for such classification.

The NRC staff should focus on the following elements of an electrical system relied upon in the design, as appropriate, and discussed in the CP application:

- a description of the utility grid and its interconnections to other grids and to the nuclear unit
- a description and configuration of the onsite alternating and direct current power systems
- a description of the methodology for coping with station blackout and the alternate alternating current power source, if provided
- design bases, criteria, standards, RGs, and technical positions that will be implemented in the design of the electric power systems

The design of the facility at the CP stage is not complete, and some calculations and analyses will be preliminary in nature and subject to change in the future. Consistent with 10 CFR 50.35, some technical and design information may reasonably be left for a later stage of licensing. However, the NRC staff must have confidence that the information to satisfy open safety questions can be resolved before completion of the construction of the facility. Examples of items that could reasonably be left for later include the following:

- for offsite power systems:
  - failure modes and effects analysis of the switchyard
  - capacity and capability of the circuits from the offsite system to the onsite distribution buses
  - provisions for grounding, surge protection, and lightning protection
  - testing the transfer of the source of power feeding the onsite distribution system
  - development of technical specifications
- for onsite alternating current power systems:
  - power system analysis studies, including load flow with voltage regulation, short circuit analysis, equipment sizing studies, protective relay setting and coordination, motor starting, grounding system design, and insulation coordination
  - testability, capacity, and capability of the onsite power system
  - reliability program for emergency onsite alternating current power sources
- for direct current power systems:
  - testability, capacity, and capability of the onsite direct current power system

Radioactive Waste Management

SRP Chapter 11, “Radioactive Waste Management,” does not provide detailed guidance to review the radioactive waste management in a CP application. The NRC staff should approach this review consistent with the SRP and the requirements in the following:

- 10 CFR 50.34a, “Design objectives for equipment to control releases of radioactive material in effluents—nuclear power reactors,” as it applies to a CP
- Appendix I, “Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion ‘As Low as is Reasonably Achievable’ for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents,” to 10 CFR Part 50 (Reference 26)
- General Design Criterion (GDC) 60, “Control of releases of radioactive materials to the environment”; GDC 61, “Fuel storage and handling and radioactive control”; GDC 63,

"Monitoring fuel and waste storage"; and GDC 64, "Monitoring radioactivity releases," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50 (Reference 27)

The staff should also consider the information that provides reasonable assurance that the applicant will comply with the requirements in 10 CFR Part 20, "Standards for protection against radiation" (Reference 28).

REFERENCES

1. 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, November 1978 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML011340122).

2. 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).

3. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.206, "Applications for Nuclear Power Plants," Revision 1, October 2018 (ADAMS Accession No. ML18131A181).

4. Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic licensing of production and utilization facilities."

5. 10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants."

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7. U.S. Nuclear Regulatory Commission, Management Directive and Handbook 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," September 20, 2019 (ADAMS Accession No. ML18093B087).

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11. 10 CFR Part 100, "Reactor site criteria."

12. 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).

13. Institute of Electrical and Electronics Engineers (IEEE) Standard 344, "IEEE Standard for Seismic Qualification of Equipment for Nuclear Power Generating Stations."

14. American Society of Mechanical Engineers, ASME QME-1, "Qualification of Active Mechanical Equipment Used in Nuclear Facilities."

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. American Concrete Institute, ACI-359, "Concrete Containments for Nuclear Reactors."

17. 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).

18. American Concrete Institute, ACI-349, "Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary."

19. American Institute of Steel Construction (AISC)/American National Standards Institute (ANSI), N690, "Specification for Safety-Related Steel Structures for Nuclear Facilities."

20. American Iron and Steel Institute, AISI S100-16, "North American Specification for the Design of Cold-Formed Steel Structural Members."

21. American Society of Mechanical Engineers, ASME AG-1, "Code on Nuclear Air and Gas Treatment."

22. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.54, "Service Level I, II, III, and In-Scope License Renewal Protective Coatings Applied to Nuclear Power Plants," Revision 3, April 2017 (ADAMS Accession No. ML17031A288).

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

24. 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."

25. U.S. Nuclear Regulatory Commission, "Design-Specific Review Standard for NuScale Small Modular Reactor Design," August 4, 2016 (ADAMS Package Accession No. ML15355A295).

26. Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents," to 10 CFR Part 50.

27. Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50.

28. 10 CFR Part 20, "Standards for protection against radiation."