DNRL-ISG-2022-01

Safety Review of Light-Water Power Reactor Construction Permit Applications

Interim Staff Guidance

October 2022

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ADAMS Accession Nos.: Package - ML22189A097; ISG –ML22189A099; Enclosure – ML22189A100; FRN –ML22189A101; CRA Summary – ML22189A098

OFFICE	NRR/DNRL/NRLB:PM	QTE	NRR/DNRL/NRLB:PM	NRR/DRO/IRAB:PM
NAME	CLauron		CLauron	CCauffman
DATE	07/19/2022	07/27/2022	8/1/2022	8/1/2022
OFFICE	NSIR/DPR: D	NRR/DNRL/NRLB:BC	NRR/DEX: D	NRR/DSS: D
NAME	KBrock	MDudek	EBenner	JDonoghue
DATE	8/2/2022	8/2/2022	8/2/2022	8/5/2022
OFFICE	NRR/DANU: D	NRR/DORL: D	NRR/DRA: D	NSIR/DPCP:D
NAME	MShams	JZimmerman for BPham	MFranovich	MSamson
DATE	8/3/2022	8/8/2022	8/8/2022	8/2/2022
OFFICE	NRR/DRO/IRAB: BC	OGC	NRR/DNRL:D	NSIR (Review Only)
NAME	PMckenna	Michael Spencer	BSmith	CRaynor
DATE	8/4/2022	9/30/2022	10/4/2022	8/1/2022

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INTERIM STAFF GUIDANCE

SAFETY REVIEW OF LIGHT-WATER POWER REACTOR

CONSTRUCTION PERMIT APPLICATIONS

DNRL-ISG-2022-01

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 and to supplement 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) (Ref. 1).

BACKGROUND

The NRC anticipates the submission of power reactor CP applications in 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" (Ref. 2), 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" (Ref. 3), using the guidance in the SRP and Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," issued June 2007 (Ref. 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 (Ref. 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, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," issued November 1978 (Ref. 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, as well as 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. (Ref. 7), and Northwest Medical Isotopes, LLC (Ref. 8). Some of the lessons learned from these reviews apply to the review of power reactor CP applications, as discussed below.

APPLICABILITY

This guidance applies to all applicants for a CP for a light-water power reactor under 10 CFR Part 50 but not to non-LWR applicants or those following the Advanced Reactor Content of Application Project (ARCAP) guidance to the extent the guidance is issued as final and is relevant to the application from a technical and regulatory perspective.

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.

Requirements for a Power Reactor Construction Permit Application

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”
- 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”

[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 50.55a, “Codes and standards”
- 10 CFR 50.150, “Aircraft impact assessment”
- 10 CFR Part 20, “Standards for Protection against Radiation” (Ref. 9)
- 10 CFR Part 100, “Reactor Site Criteria” (Ref. 10)

The following discussion elaborates on certain CP requirements.

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 an explanation of how the design bases relate to the principal design criteria, and information on the materials of construction, general arrangement, and approximate dimensions sufficient for the staff to conclude that the final design will conform to the design bases with an adequate margin for safety. In accordance with 10 CFR 50.34(a)(1)(ii), the application must provide a description and safety assessment of the site and a safety assessment of the facility, and the Commission expects that reactors will reflect through their design, construction, and operation an extremely low probability for accidents that could result in the release of significant quantities of radioactive fission products.

The regulations in 10 CFR 50.34a require a description of the preliminary design of equipment to maintain control of radioactive material in effluents produced during normal reactor operations, and of the design objectives and means for keeping the levels of radioactive material in effluents as low as is reasonably achievable. Furthermore, 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 and to describe the provisions for packaging and storing radioactive solid waste materials and shipping them off site.

The regulations in 10 CFR 50.150 require CP applicants to perform a realistic design-specific assessment of how the impact of a large commercial aircraft would affect the facility and to identify and incorporate into the design those design features and functional capabilities that 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. Note that 10 CFR 50.150 requires applicants to perform aircraft impact assessments at both the CP and OL stages and include the required information in both applications, based on the level of design information available at the time of each application. 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 an impact assessment, which is not required to be submitted to the NRC (74 FR 21820; June 12, 2009) (Ref. 11).

Issuance of a Construction Permit

The NRC may issue the CP if the agency makes the findings listed in 10 CFR 50.35(a). Pursuant to 10 CFR 50.35(b), a CP authorizes the applicant to proceed with construction but will not constitute Commission approval of the safety of any design feature or specification unless the applicant specifically requests and receives such approval and such approval is incorporated in the permit. While 10 CFR 50.35 provides some flexibilities for applicants, this does 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, issued a "provisional" CP when an applicant had not submitted all the technical information necessary 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 Atomic Energy Commission proposed codifying the Commission's practice for issuing a CP (34 FR 6540; April 16, 1969) (Ref. 12). 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" CPs (35 FR 5317; March 31, 1970) (Ref. 13). 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.[2]

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.
>
> Note:
> When an applicant has supplied initially all of the technical information required to complete the application, including the final design of the facility, the findings required above will be appropriately modified to reflect that fact.

[2] An example is the CP issued for the Shearon Harris Nuclear Power Plant (Ref. 14). CPs also included permit conditions on specified issues.

If a novel design has not sufficiently progressed and certain information is not available at the time of CP application submittal, 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 providing 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 public health and safety. The CP application must address all regulatory requirements applicable to a CP.

The required findings in 10 CFR 50.35(a) focus on the safety aspects of the design, including the principal architectural and engineering criteria and safety design features, and siting information to support construction of the facility. As 10 CFR 50.35(a) states, these findings were written for an application that does not contain sufficient information for the NRC to approve all proposed design features. Given the technological advances since the most recent amendment of the regulation, an applicant may provide more complete technical information in its CP application than was historically presented and thereby reduce the regulatory review in the subsequent OL review phase. As noted in 10 CFR 50.35(a), if specifically requested by the applicant, the findings in 10 CFR 50.35(a) will be modified for a complete CP application that provides 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.[3] Any 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 to be available to support a COL application. Any approval, if granted, would apply only to the extent that the item is fully addressed or treated in the application and would not extend beyond items or details not fully covered therein. 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 any 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 in accordance with the requirements of the license and NRC regulations 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. If the NRC determines that all applicable requirements are met, the Commission will issue an OL permitting the applicant to 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

[3] 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 feedback on and staff review of unique design aspects of the facility, and preparing resources for the application review.

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 LWR designs will comply with the Commission's regulations and adequately protect public health and safety.

The NRC staff should review the risk- and safety-significant aspects of the application commensurate with their level of 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 it 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." The staff requirements memorandum (SRM) for SECY-98-144, "Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation," dated March 1, 1999 (Ref. 15), 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 using 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 to 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 concerns 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 included in applications, and provides insights on what information in an application would support the NRC staff's safety review and evaluation.

The initial issuance of RG 1.206 (Ref. 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 (Ref. 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 to be included in the safety analysis report, which is addressed 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 (Ref. 16), on using, in appropriate circumstances, the same reasoned decision-making process 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 29, 2018 (Ref. 17), and apply the principles of good regulation discussed in the memorandum from Ho Nieh dated October 15, 2019 (Ref. 18).

<u>Special Topics</u>

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's 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) (Ref. 19), 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 review that begins with the design, construction, and operating features described in the applicant's PSAR. The plant design and operating features may be preliminary when construction begins, with NRC evaluation of the final design, including the FSAR-level of design detail describing the facility as constructed, 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 construction is completed.

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 early feedback on and staff review of issues associated with the 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 particular facility designs.

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 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 relevant to LWR license applicants and, if fully executed, will enable the NRC staff to offer more predictable and shorter schedules and other benefits when reviewing a reactor license application.

Consistent with regulatory requirements and Commission policy statements, the NRC staff is more fully integrating the use of risk insights into preapplication activities by aligning its review focus and resources to risk-significant structures, systems, and components (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, in February 2016, and (2) Northwest Medical Isotopes, in May 2018. The NPUF lessons learned, which are described below, may improve the effectiveness and efficiency of safety reviews of PSARs to determine whether an 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 descriptions for programs implemented during operation.

- If the PSAR includes preliminary descriptions of the facility's 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. 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 periodic reports on such information to the NRC before construction is completed.

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, related to the definition of construction and LWAs.

Questions have been raised about the possibility of submitting an OL application before the NRC issues a CP. The NRC staff is still considering the legal, policy, and timing implications of this action. For OL applications submitted before a 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 submittal of the CP application. In such a case, addressing this reclassification would result in an extensive reworking 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 finally resolve matters within their scopes 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 satisfies the standards for referencing a 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 departures or variances 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 previously approved by the NRC.

For a CP application referencing an ESP, the NRC staff's review and evaluation would include 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, 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. Furthermore, 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 NRC discusses the basis for this restriction in the final rule for the U.S. Advanced Boiling Water Reactor DC (62 FR 25800; May 12, 1997) (Ref. 21).

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 departures or variances 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, "Staff Requirements—SECY-15-0002—Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications," dated September 22, 2015 (Ref. 22). This 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, as a CP applicant must comply with the applicable regulations in effect at the time the NRC issues the CP.

In 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) (Ref. 20) and other Commission direction identified in SECY-15-0002, dated January 8, 2015 (Ref. 23), 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 this policy statement, an applicant submitting a new design for NRC approval could address severe accidents acceptably if it does the following:

- Demonstrates compliance with the procedural requirements and criteria of the current Commission regulations, including the Three Mile Island requirements described in 10 CFR 50.34(f).

- Demonstrates 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 electrical supply systems.

- Completes a PRA and consideration of the severe accident vulnerabilities the PRA exposes, along with the insights that the PRA may add to the assurance of no undue risk to public health and safety.

- Completes a staff review of the design with a conclusion of safety acceptability using an approach that stresses deterministic engineering analysis and judgment complemented by the PRA.

The staff discusses its proposal to apply this policy statement and other Commission direction to new 10 CFR Part 50 power reactor applications in SECY-15-0002. The other Commission direction discussed in that SECY includes the Commission direction in response to SECY-89-013, “Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs),” dated January 19, 1989 (Ref. 24); SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” dated January 12, 1990 (Ref. 25); and SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated April 2, 1993 (Ref. 26).

Through the ARCAP, the NRC is working to develop technology-inclusive, risk-informed, and performance-based application guidance. This ARCAP guidance is intended for use by advanced reactor applicants 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. When final, the NRC staff may consider whether the ARCAP guidance is relevant to reviews of LWR CP applications.

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” (Ref. 27); 10 CFR Part 40, “Domestic Licensing of Source Material” (Ref. 28) ; or 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material” (Ref. 29). 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, which was the historical practice.

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, ISSUE FINALITY, AND FORWARD FITTING DISCUSSION

This ISG provides guidance for the NRC staff review of light-water power reactor construction permit applications. Issuance of this final ISG would not constitute backfitting as defined in 10 CFR 50.109 (the Backfit Rule) and as described in NRC Management Directive 8.4; would not affect the issue finality of an approval under 10 CFR part 52; and would not constitute forward fitting as that term is defined and described in Management Directive 8.4.

The NRC staff's position is based upon the following considerations:

- The final ISG positions would not constitute backfitting or forward fitting or affect issue finality, inasmuch as the ISG would be internal guidance to NRC staff. The ISG provides interim guidance to the staff on how to review an application for NRC regulatory approval in the form of licensing. Changes in internal staff guidance, without further NRC action, are not matters that meet the definition of backfitting or forward fitting or affect the issue finality of a part 52 approval.

- Backfitting and issue finality—with certain exceptions discussed in this section—do not apply to current or future CP applicants. CP applicants and potential CP applicants are not, with certain exceptions, the subject of either the Backfit Rule or any issue finality provisions under 10 CFR Part 52. This is because neither the Backfit Rule nor the issue finality provisions of 10 CFR Part 52 were intended to apply to every NRC action that substantially changes the expectations of current and future applicants. The exceptions to the general principle, as applicable to guidance for CP applications, are whenever a 10 CFR Part 50 CP applicant references a license (e.g., an early site permit) or an NRC regulatory approval (e.g., a design certification rule) (or both) for which specified issue finality provisions apply. The NRC staff does not currently intend to impose the positions represented in this ISG in a manner that constitutes backfitting or is inconsistent with any issue finality provision of 10 CFR Part 52. If in the future the NRC staff seeks to impose positions stated in this ISG in a manner that would constitute backfitting or be inconsistent with these issue finality provisions, the NRC staff must make the requisite showing as set forth in the Backfit Rule or address the regulatory criteria set forth in the applicable issue finality provision, as applicable, that would allow the staff to impose the position.

- Forward fitting—The Commission's forward fitting policy generally does not apply when an applicant files an initial licensing action for a new facility. Nevertheless, the staff does not, at this time, intend to impose the positions represented in the final ISG in a manner that would constitute forward fitting.

CONGRESSIONAL REVIEW ACT

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

FINAL RESOLUTION

The staff will transfer 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 transfer of all pertinent information and guidance in this document into the SRP, this ISG will be closed.

REFERENCES

1. 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)

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

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

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

5. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.206, Revision 1, "Applications for Nuclear Power Plants," October 2018 (ML18131A181).

6. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," November 1978 (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 (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 (ML18037A468).

9. 10 CFR Part 20, "Standards for Protection Against Radiation."

10. 10 CFR Part 100, "Reactor Site Criteria."

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

12. "Backfitting of Production and Utilization Facilities; Construction Permits and Operating Licenses," 34 FR 6540; April 16, 1969 (proposed rule).

13. "Backfitting of Production and Utilization Facilities; Construction Permits and Operating Licenses," 35 FR 5317; March 31, 1970 (final rule).

14. U.S. Nuclear Regulatory Commission, Correspondence from Roger S. Boyd, "Issuance of Construction Permits—Shearon Harris Nuclear Power Plant, Units 1, 2, 3 and 4," January 27, 1978 (ML020560123).

15. U.S. Nuclear Regulatory Commission, SRM-SECY-98-144, “Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation,” March 1, 1999 (ML003753601).

16. U.S. Nuclear Regulatory Commission, Management Directive and Handbook 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests,” September 20, 2019 (ML18093B087).

17. U.S. Nuclear Regulatory Commission, Memorandum from Frederick Brown, “Expectations for New Reactor Reviews,” August 29, 2018 (ML18240A410).

18. U.S. Nuclear Regulatory Commission, Memorandum from Ho Nieh, “Applying the Principles of Good Regulation as a Risk-Informed Regulator,” October 15, 2019 (ML19260E683).

19. “Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Reactors,” 53 FR 32060; August 23, 1988 (proposed rule).

20. “Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants,” 50 FR 32138, August 8, 1985 (Severe Accident Policy Statement).

21. “Standard Design Certification for the U.S. Advanced Boiling Water Reactor Design,” 62 FR 25800; May 12, 1997 (final rule).

22. U.S. Nuclear Regulatory Commission, 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 (ML15266A023).

23. U.S. Nuclear Regulatory Commission, SECY-15-0002, “Proposed Updates of Licensing Policies, Rules, and Guidance for Future New Reactor Applications,” January 8, 2015 (ML13277A420).

24. U.S. Nuclear Regulatory Commission, SECY-89-013, “Design Requirements Related to the Evolutionary Advanced Light Water Reactors (ALWRs),” dated January 19, 1989 (ML003707947).

25. U.S. Nuclear Regulatory Commission, SECY-90-016, “Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements,” dated January 12, 1990 (ML003707849).

26. U.S. Nuclear Regulatory Commission, SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs,” dated April 2, 1993 (ML003708021).

27. 10 CFR Part 30, “Rules of General Applicability to Domestic Licensing of Byproduct Material.”

28. 10 CFR Part 40, “Domestic Licensing of Source Material.”

29. 10 CFR Part 70, “Domestic Licensing of Special Nuclear Material.”

APPENDIX A

CLARIFICATIONS TO THE EXISTING REVIEW GUIDANCE IN NUREG-0800

An applicant may use the information in Regulatory Guide (RG) 1.70, Revision 3, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants: [Light-Water Reactor (LWR)] Edition,” issued November 1978 (Ref. 1); and RG 1.206, Revision 1, “Applications for Nuclear Power Plants,” issued October 2018 (Ref. 2), 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” (Ref. 3), or 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants” (Ref. 4). 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 construction permit (CP) application structure consistent with NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (SRP) (Ref. 5); helps to ensure the completeness of the information in the application; and provides insights on the information that should be included in applications 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 interim staff guidance, and be aware that recent updates to the SRP focused on guidance to support the review of combined license 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’s policy in Management Directive 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests,” dated September 20, 2019 (Ref. 6), 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 29, 2018 (Ref. 7), and apply the principles of good regulation discussed in the memorandum from Ho Nieh dated October 15, 2019 (Ref. 8).

The NRC staff should review risk-significant and safety-significant aspects of the application commensurate with their level of 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 or safety significance. An aspect of a design can be more or less risk-significant than it 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) (Ref. 9), 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." The staff requirements memorandum on SECY-98-144, "Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation," dated March 1, 1999 (Ref. 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 decisions.

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, "Conduct of Operations," includes guidance on the requirements of 10 CFR Part 100, "Reactor Site Criteria" (Ref. 11), related to the development of security and emergency plans. The NRC expects that the applicant will completely characterize the site selected for construction. Furthermore, 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 structures, systems, and components (SSCs) unless the applicant requests 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 be confident 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. 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.

Reviewing transient and accident analyses requires an evaluation of analytical methods, inputs, and results of analyses. In most cases, applications do not document analytical methods but instead refer to a vendor topical report. Examples of such methods for 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 for the preliminary design using a detailed quantitative analysis. Note that each category may have multiple limiting events to the various acceptance criteria based on different assumed initial conditions and equipment behavior. The preliminary safety analysis report (PSAR) should include the rationale for the determination that the limiting event is in fact limiting. For nonlimiting events within a category, the NRC staff verifies that the PSAR documents the rationale for this determination; in such cases, quantitative or qualitative technical justification may be acceptable. 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 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 that there is reasonable assurance that 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, 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 be confident that any missing information and open safety questions can be resolved satisfactorily before the completion of facility construction. Examples of items that could reasonably be left for later include the following:

- Analyses for regulated beyond-design-basis events (e.g., station blackout, anticipated transients without scram, mitigation strategies for beyond-design-basis external events).

- 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.

- Finalization 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, (Ref. 12) 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:

  – The latest revision of RG 1.100, "Seismic Qualification of Electrical and Active Mechanical Equipment and Functional Qualification of Active Mechanical Equipment for Nuclear Power Plants," (Ref. 13) for the qualification of mechanical and electrical equipment.

  – The latest revision of RG 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments," (Ref. 14), for the design and qualification of concrete containment.

  – The latest NRC-endorsed edition of the American Institute of Steel Construction (AISC)/American National Standards Institute (ANSI) N-690, "Specification for Steel Related Structures for Nuclear Facilities," for the design of safety-related steel structures and the design of the spent fuel pool liner (Ref. 17).

  – The latest revision of RG 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)," (Ref. 15), for the design and qualification of the safety-related concrete structures other than containment.

  – The latest revision of RG 1.199, "Anchoring Components and Structural Supports in Concrete," (Ref. 16).

- For the cold-formed support members of conduit and cable trays, the latest NRC-endorsed revision of the American Iron and Steel Institute (AISI) S100-16, "North American Specification for the Design of Cold-Formed Steel Structural Members" (Ref. 18), is acceptable. For the hot-rolled support members of conduit and cable trays, the latest NRC-endorsed revision of AISC/ANSI N690 is acceptable.

- The staff should review the general construction of ducts including safety-related heating, ventilation, and air conditioning (HVAC) ductwork. For HVAC cold-formed member supports, the latest NRC-endorsed revision of AISI S100-16 is acceptable. For hot-rolled structural members of the HVAC supports, the latest NRC-endorsed revision of 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 containment areas 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 and two American Society for Testing and Materials (ASTM) standards referenced in the RG. The current revision of RG 1.54 is Revision 3, "Service Level I, II, III, and In Scope License Renewal Protective Coatings Applied to Nuclear Power Plants," issued April 2017 (Ref. 19), which references ASTM D5144, "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants" (Ref. 20), and ASTM D3911, "Standard Test Method for Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design Basis Accident (DBA) Conditions" (Ref. 21). 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 the revision of ASTM D3911 referenced in the RG 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 the revision of ASTM D5144 referenced in the RG 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 (Ref. 22) 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. The guidance in NuScale Chapter 7 DSRS reflects an approach that a prospective applicant may use to develop a unifying I&C

framework that addresses all the significant aspects of the I&C design in a unified manner to minimize the repetition of the requirements.

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, 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 and discussed in the CP application, as applicable:

- 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 be confident that the information to satisfy open safety questions can be resolved before the completion of facility construction. 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.
  - Finalization 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.
- 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.

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" (Ref. 23).

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2. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.206, Revision 1, "Applications for Nuclear Power Plants," October 2018 (ML18131A181).

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

4. 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants."

5. 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)

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8. U.S. Nuclear Regulatory Commission, Memorandum from Ho Nieh, "Applying the Principles of Good Regulation as a Risk-Informed Regulator," October 15, 2019 (ML19260E683).

9. "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," (Probabilistic Risk Assessment (PRA) Policy Statement), Volume 60 of the *Federal Register* (FR), page 42622 (60 FR 42622), August 16, 1995.

10. U.S. Nuclear Regulatory Commission, SRM-SECY-98-144, "Staff Requirements—SECY-98-144—White Paper on Risk-Informed and Performance-Based Regulation," March 1, 1999 (ML003753601).

11. 10 CFR Part 100, "Reactor Site Criteria."

12. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (ASME Code), Section III, "Rules for Construction of Nuclear Facility Components," Division 1, "Metallic Components."

13. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.100, "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants."

14. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.136, "Design Limits, Loading Combinations, Materials, Construction, and Testing of Concrete Containments."

15. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.142, "Safety-Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments)."

16. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.199, "Anchoring Components and Structural Supports in Concrete."

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

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