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Content of Risk Assessment and Severe Accident Information in Light-Water Power Reactor Construction Permit Applications

Interim Staff Guidance

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INTERIM STAFF GUIDANCE

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PURPOSE

This interim staff guidance (ISG) clarifies the scope and depth of the staff review of the content of risk assessment and severe accident information in a construction permit (CP) application for a light-water power reactor. It supplements the guidance in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition (NUREG-0800, Formerly issued as NUREG-75-087)," ("the SRP"; Ref. 1).[1]

BACKGROUND

The review of an application for a CP to build a light-water reactor (LWR) 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). This process involves the issuance of a CP based on preliminary design information documented in a preliminary safety analysis report (PSAR), which allows an applicant to begin construction. After construction is essentially complete, the licensee will supply a final safety analysis report (FSAR) with the application for an operating license (OL). The FSAR should describe the complete and final design of the facility as constructed; identify the changes from the criteria, design, and bases in the PSAR; and discuss the bases for and safety significance of the changes from the PSAR. The U.S. Nuclear Regulatory Commission (NRC) has not issued a power reactor CP since the 1970s.

More recently, the NRC has issued combined licenses (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. The NRC issued guidance to applicants for preparing COL applications in 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 RG 1.206, Revision 1, "Applications for Nuclear Power Plants," in October 2018 (Ref. 5).

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 acceptable for PSARs in CP applications. However, this guidance has not been updated since 1978 and these insights may be limited to the degree that the guidance does not account for subsequent requirements, NRC technical positions, novel design approaches, or advances in technical knowledge.

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1 The SRP contains review guidance for an application to build and operate an LWR, whether the application is submitted under 10 CFR Part 50 or 10 CFR Part 52.

On October 31, 2022, the NRC staff issued ISG DNRL-ISG-2022-01, "Safety Review of Light-Water Power Reactor Construction Permit Applications" (Ref. 7), to facilitate safety reviews of LWR CP applications and to supplement the guidance in the SRP. DNRL-ISG-2022-01 describes the regulatory requirements, applicable review guidance in the SRP, and special topics for an LWR CP application. DNRL-ISG-2022-01 provides guidance on the staff review of the preliminary design information in the PSAR including the description and safety assessment of the site on which the facility is to be located. DNRL-ISG-2022-01 does not provide specific information relevant to the review of probabilistic risk assessment (PRA) and alternative risk evaluations supporting an LWR CP application but points generally to the SRP to provide the NRC staff with an acceptable approach for reviewing such information. As stated above, the staff has developed this draft ISG to clarify the scope and depth of the staff review of the content of risk assessment and severe accident information in a CP application for a light water power reactor.

The staff has engaged with stakeholders in several public meetings on this topic; the staff considered stakeholder views stated in these meetings in formulating the positions presented in this ISG.[2] This ISG replaces the NRC staff-developed draft white paper, dated November 29, 2023 (Ref. 8) that was not issued as an official agency position on this subject, and is available only as historical background information.

RATIONALE

An applicant for a CP is required to include a PSAR in its CP application under 10 CFR 50.34, "Content of applications; technical information," which also identifies the minimum information to be included in the PSAR.

The technology of risk assessment and severe accident analyses has advanced significantly since the last power reactor CP was issued in 1978. Licensees and the NRC staff use risk assessment techniques more effectively than ever before. In addition, designers of new LWRs are using risk assessment, including formal PRAs and analyses of severe accidents, to support risk-informed design decisions. A systematic approach to assessing the plant risk, including a PRA, can help demonstrate that the application complies with the regulations and follows Commission policy. This includes, but is not limited to—

(1) meeting 10 CFR 50.34(a)(1)(ii), under which reactors are expected to 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

(2) comparing assessed risk against the quantitative health objectives as stated in Commission Policy

(3) achieving the Commission's policy goals for containment performance

(4) identifying severe accident vulnerabilities and corresponding design improvements

(5) determining licensing-basis events

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[2] Meeting summaries can be found using the following Agencywide Documents Access and Management System (ADAMS) Accession Numbers: ML23104A314 (initial public meeting), ML23243A010 (second public meeting), and ML24047A232 (final public meeting).

(6) supporting classification of structures, systems, and components (SSCs), including identification of non-safety-related systems that need regulatory oversight

(7) supporting the adequacy of the plant's defense-in-depth capability

Under 10 CFR 50.35, "Issuance of construction permits," a CP application may be submitted even if it does not initially supply all the technical information required to support approval of all proposed design features. Under such circumstances, the Commission may issue a CP, provided the findings in 10 CFR 50.35(a) can be made.

If an applicant chooses to rely on the results and insights from its PRAs and alternative risk evaluations described in its CP application,[3] the staff should generally have confidence in the following items in order to rely on those results and insights to make the findings required under 10 CFR 50.35(a).

(1) In combination with submitted design information, supplemental analyses, and commitments, PRAs and alternative risk evaluations possess the characteristics, attributes, and capabilities needed to provide results and insights as bases for design decisions. The results and insights will support the development of a PRA to support an OL application, including the confirmation of changes made during construction, including changes in the design as described in the CP application.

(2) PRAs and alternative risk evaluations used in support of the CP application are reasonably consistent with the maturity and completeness of the design information submitted. Accordingly, PRAs and alternative risk evaluations appropriately represent each modeled hazard, the plant's response to upset conditions caused by these hazards, and the plant's capacity to withstand the hazards.

(3) The results and insights from PRAs and alternative risk evaluations are reasonable. The CP application identifies how the PRA and alternative risk evaluations are used to support or confirm design and licensing decisions.

(4) The use of the PRAs and alternative risk evaluations address relevant Commission policies, including, but not limited to, searching for severe accident vulnerabilities and meeting the Commission's safety goals.

This confidence informs the staff's evaluation of the CP application for the purpose of determining whether the findings under 10 CFR 50.35(a) can be made.

Another important use of the PRA and alternative risk evaluations in the CP application is to focus the NRC staff's review on those aspects of the design that contribute most to safety and minimize attention to issues of low risk or low safety significance. Consistent with the NRC's use

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[3] The term "alternative risk evaluation" is intended to encompass a range of approaches. These are not considered to be PRA approaches as defined in RG 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Ref. 10). Examples of alternative risk evaluations that may be considered for a CP application are listed in Table 2, "Additional Elements to Support a CP Application," below.

of risk-informed decision-making, the NRC staff should integrate risk insights with traditional engineering approaches when making regulatory decisions on a CP application.

APPLICABILITY

This guidance applies to the review of all CP applications for a light-water power reactor under 10 CFR Part 50.

GUIDANCE

This document provides guidance to the staff on the acceptability of the description of the PRA and its results and severe accident information in PSARs. Specifically, it addresses PRAs, alternative risk evaluations, and severe accident analyses relied upon to make design and licensing decisions on an LWR CP application, and the specific regulatory findings made under 10 CFR 50.35(a). In doing this, the staff should be able to identify design-basis events, design features to address severe accident vulnerabilities, and how the applicant demonstrates conformance to relevant Commission policy (e.g., safety goals).

The guidance contained in this document is focused on CP applications for LWR designs that do not use the licensing modernization project (LMP) framework.[4] It considers the role of PRAs and alternative risk evaluations, the severe accident analysis at the time an application is submitted, and the flexibility intended to be afforded by the two-step licensing process under 10 CFR Part 50. The scope and technical acceptability of the CP application PRA depend on the intended use of the information and the level of design maturity. The information identified in this guidance for PRAs and alternative risk evaluations in an LWR CP application addresses relevant Commission policies and key industry and NRC guidance documents on the use of PRAs in support of regulatory decision-making.

RG 1.200, combined with DC/COL-ISG-028, “Assessing the Technical Adequacy of the Advanced Light-Water-Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application: Interim Staff Guidance,” dated December 2, 2016 (Ref. 11), provides staff positions on determining whether a design-specific or plant-specific PRA used to support a 10 CFR Part 52 LWR application is sufficient to provide confidence in the results for regulatory decision-making. RG 1.200 applies to the full scope of risk contributors considered by PRAs and for a plant’s entire life cycle. With RG 1.200, DC/COL-ISG-028 provides staff positions and clarifications on supporting requirements in American Society of Mechanical Engineers/American Nuclear Society (PRA Standard) (ASME/ANS) RA-

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[4] The NRC endorsed the LMP methodology in RG 1.233, “Guidance for a Technology-Inclusive, Risk-Informed, and Performance-Based Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors” (Ref. 12). RG 1.233 endorses the LMP methodology, as it is described in Nuclear Energy Institute (NEI) 18-04, Revision 1, “Risk-Informed Performance-Based Technology Inclusive Guidance for Non-Light-Water Reactor Licensing Basis Development,” issued August 2019 (Ref. 13), but only for Non-LWR applications.

Guidance on the content of non-LWR applications using the LMP methodology can be found in NEI 21-07, Revision 1, “Technology Inclusive Guidance for Non-Light Water Reactors; Safety Analysis Report Content for Applicants Using the NEI 18-04 Methodology” (Ref. 14), as endorsed in RG 1.253, “Guidance for a Technology-Inclusive Content of Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactors” (March 2024).

Sa-2009, “Addenda To ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” that are not applicable or cannot be achieved as written for the DC and COL application stages. Staff should consider this approach in determining the applicability of supporting requirements in industry standards to a CP application for an LWR. DC/COL-ISG-028 is therefore one example of the results of the application process described in the PRA Standard and endorsed in RG 1.200 to determine whether every supporting requirement (SR) is needed for a high-level requirement.

Applicable Regulations, Commission Policy Statements, and Guidance Documents

The key regulations relevant to the scope of this guidance development effort are 10 CFR 50.34(a) and 10 CFR 50.35(a).

The regulations in 10 CFR 50.34(a) set requirements for the content of CP applications, including the substance of the PSAR that must be submitted as part of the application. As discussed in DNRL-ISG-2022-01, in accordance with 10 CFR 50.34(a)(1)(ii), the CP application must provide a description and safety assessment of the site and a safety assessment of the facility. As stated in 10 CFR 50.34(a)(1)(ii), 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 regulation in 10 CFR 50.35(a) specifies the findings necessary for the Commission to issue a CP. If there are design features that can reasonably be left for later consideration or for which final approval is not sought, the applicant will have to supply the additional technical or design information needed to complete the safety analysis. The final safety analysis report required with the OL application must include this information. DNRL-ISG-2022-01 provides additional information on meeting the requirements under 10 CFR 50.35(a).

The CP application must describe safety features or components that require research and development. In such cases, the staff should verify that the application includes a description of a research and development program that will be conducted to resolve any safety questions associated with such features or components.

Based on these items, the staff should determine whether there is reasonable assurance that safety questions requiring research and development will be satisfactorily resolved before construction is completed. In order for the NRC to issue a CP, the staff must find there is reasonable assurance that the plant can be constructed and operated at the proposed location without undue risk to the health and safety of the public. In cases where an applicant has initially supplied all the technical information required to support the issuance of a CP that approves all design features, the findings will reflect that all design features were approved.

As described in 10 CFR 50.35(b), an applicant may request Commission approval of the safety of a design feature or specification in a CP application. When the application includes a safety approval request, the staff should ensure that additional information has been provided beyond that identified in this document and that information is sufficient to demonstrate the acceptability of the request. Such information will normally be consistent with the type and level of detail of information provided at the OL stage. For such cases, the PRA acceptability should be generally consistent with that for a combined license applicant, as discussed in chapter 19 of the SRP, RG 1.200, and DC/COL-ISG-028, and is not discussed further in this guidance. PRAs and alternative risk evaluations may be used in an LWR CP application to support meeting specific regulations such as 10 CFR 50.34(f)(1)(xii), (f)(2)(ix), and (f)(3)(v); 10 CFR 50.44, "Combustible gas control for nuclear reactors"; and 10 CFR 50.150, "Aircraft impact assessment." However, such uses of PRAs and alternative risk evaluations will be evaluated on a case-by-case basis. The NRC staff strongly recommends preapplication engagement for such cases.

Commission policy statements and staff requirements memorandums (SRMs) that apply to an LWR CP application include, but are not limited to, the following:

- "Policy Statement on the Regulation of Advanced Reactors," dated October 14, 2008 (Ref. 15)

- "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," dated August 8, 1985 (Ref. 16)

- "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement; Correction and Republication," dated August 21, 1986 (Ref. 17)

- "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," dated August 16, 1995 (Ref. 18)

- Staff Requirements Memorandum (SRM)-SECY-90-0016, "Staff Requirements-SECY-90-0016—Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," dated June 26, 1990 (Ref. 19)

- SRM-SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated July 21, 1993 (Ref. 20)

- SRM-SECY-94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems," dated June 30, 1994 (Ref. 21)

- SRM-SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," dated June 28, 1995 (Ref. 22)

- SRM-SECY-12-0081, "Staff Requirements-SECY-12-0081, Risk-Informed Regulatory Framework for New Reactors," dated October 22, 2012 (Ref. 23)

- 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. 9)

Guidance documents that can be applied to an LWR CP application include, but are not limited to, the following:

- DNRL-ISG-2022-01, "Safety Review of Light-Water Power Reactor Construction Permit Applications," dated October 31, 2022 (Ref. 7)

- Regulatory Guide 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities” (Ref. 10)

- DC/COL-ISG-028, “Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application: Interim Staff Guidance,” dated December 2, 2016 (Ref. 11)

- ASME/ANS RA-Sa-2009, “Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications” (Ref. 24)

- Chapter 19, “Severe Accidents,” of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (Ref. 1)

Uses of PRA Information in a Construction Permit Application

Results and insights from PRAs and alternative risk evaluations are one aspect of the overall decision-making process for making findings under 10 CFR 50.35(a); this information should not constitute the sole basis for the staff’s findings. The staff should make the overall regulatory findings in an integrated manner that considers the uses of the PRAs and alternative risk evaluations with other traditional engineering analysis tools and methods. Specifically, the staff will use the PRA and alternative risk evaluation information identified in this guidance to confirm that the CP applicant—

- identifies the uses of PRA and alternative risk evaluation insights (e.g., selection of licensing-basis-events, determination of non-safety-related systems subject to regulatory treatment, demonstration of margins to the Commission’s safety goals)

- establishes a process for identifying and incorporating into the design or construction key contributors to plant risk and risk insights from PRAs and alternative risk evaluations, consistent with their identified uses

- establishes and implements a process to systematically identify all applicable hazards and initiating events, radiological sources, and plant operating states (POSs) that need to be considered in the PRA and alternative risk evaluations (i.e., whether they are screened out or explicitly modeled) during the design and construction of the plant

- defines the metrics (e.g. core damage, large release) used to characterize plant risk

- establishes a systematic process for identifying and dispositioning uncertainties in the PRA and alternative risk evaluations (i.e., modeling, parametric, and completeness), including treatment of key assumptions and sources of uncertainty

- identifies, consistent with the most current design information, the limitations of the PRA and alternative risk evaluations supporting the description and results included in the CP application in terms of scope, level of detail, conformance with PRA technical elements, and plant representation; identifies the impact of these limitations on the results and insights; and develops a plan for addressing these limitations at the CP stage or resolving these limitations at the OL stage

- establishes a plan to control configuration management of the PRA and alternative risk evaluations during construction, including any design modifications

- identifies all methods, approaches, and standards used in the development of the PRA and alternative risk evaluations or that will be used at the OL stage, including the self-assessment and peer review processes

The staff's confirmation of the completion of these actions provides confidence that:

(1) The PRA and alternative risk evaluations and their results reflect the design described in the CP application and are reasonable.

(2) Based on the relevant commitments in the CP application and the PRA configuration control program, the PRA and alternative risk evaluations will be updated to reflect the final design and possess the minimum characteristics, attributes, and capabilities needed to support an OL application.

For cases in which the PRA and alternative risk evaluations supporting a CP application do not address all the relevant risk contributors and the applicant has made commitments essential to addressing these contributors at the OL stage of the licensing process, the staff's review involves judgment on qualitative and quantitative information presented in the PSAR, as well as the applicant's commitments.

Minimum Scope of PRA and Alternative Risk Evaluations for a CP Application

The staff should ensure that the applicant has evaluated all hazards for their impact on the risk from the design. The staff should verify that the applicant developed a full-power reactor internal events PRA for the CP application, commensurate with the maturity of the design. For the CP application, consistent with DC/COL-ISG-028, Capability Category I of an NRC endorsed PRA standard is acceptable for PRAs, including the internal events PRA.

Alternative risk evaluations for hazards, other than internal events that cannot be screened out, are acceptable for the CP application. Examples of alternative risk evaluations include PRA based seismic margin assessments and conservative assessments of non-seismic external hazards. The staff should confirm that these alternative risk evaluations incorporate site-specific information.

Tables 1 and 2 summarize the minimum scope of PRA and alternative risk evaluations for a CP application.

The staff should verify that the PRA results are quantified in terms of the risk metrics—core damage frequency (CDF), large release frequency (LRF) or large early release frequency (LERF), and conditional containment failure probability (CCFP)—in conformance with the Commission's safety goals. The staff should review the justification for alternatives, such as deterministic demonstration of containment performance in lieu of CCFP, on a case-by-case basis. SRM-SECY-90-016 discusses the applicability of the CDF and LRF to advanced LWRs licensed under 10 CFR Part 52. SRM-SECY-12-0081 approves the staff's recommendation to transition from LRF to large early release frequency (LERF) at or before initial fuel load and discontinue regulatory use of LRF and CCFP thereafter. For 10 CFR Part 50 plants, the CP PRA should use LRF or LERF and the OL PRA should use LERF (because an OL authorizes the loading of fuel) consistent with the guidance in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed-Decisions on Plant-Specific Changes to the Licensing Basis," (Ref 28). A transition between the use of LRF and CCFP to LERF at or before initial fuel load is consistent with SRM-SECY-12-0081. The use of LRF is an acceptable metric for the CP PRA because the information needed to calculate LERF may not be available.

Table 1 Internal Events PRA Elements for a CP Application*

Initiating Event Analysis
Accident Sequence Analysis
Success Criteria Development
Systems Analysis
Human Reliability Analysis
Data Analysis
Large Release Frequency Analysis†
Quantification
Uncertainty Analysis

* Capability Category I is acceptable for a CP application
† Level 2 PRA

Table 2 Additional Elements for CP Application

PRA Evaluations	Alternative Risk Evaluations (not PRA)
Internal Flood PRA	Internal flood risk evaluation
Internal Fire PRA	Internal fire risk evaluation
Seismic PRA	PRA -based seismic margins
High-winds PRA or PRA -based screening analysis	High-winds risk evaluation
External flooding PRA or PRA -based screening analysis	External flood risk evaluation
Other hazards PRA or PRA -based screening analysis	Other hazards risk evaluations
Low-power and shutdown PRA	Low-power and shutdown risk evaluation
	Plant operating state analysis

Submittal Information for the PRA and Alternative Risk Evaluations in a CP Application

The reviewer should confirm that the applicant's PSAR demonstrates that the fundamental design and operation of the plant have been accurately represented in the PRA and alternative risk evaluations supporting the CP application, and that the development of the PRA and alternative risk evaluations has been successfully executed for the CP application. One way to demonstrate this for PRAs is to determine whether the PRA meets the staff positions in RG 1.200 and DC/COL-ISG-028 as they relate to the foundational PRA elements, which are essential for a base PRA. Such a demonstration should be used to establish confidence in the applicant's technical qualifications in developing the PRA and in the use of the resulting risk insights.

The reviewer should ensure that the CP application includes the following information on the uses of the PRA as input to the regulatory findings:

- a discussion of all the uses of the CP application PRA and alternative risk evaluations and resulting risk insights (e.g., identification of severe accident vulnerabilities, identification of design options to reduce risk, selection of licensing basis events, determination of non-safety-related systems subject to regulatory treatment, demonstration of margins to the Commission's safety goals)

- identification of design options to reduce risk and address severe accident vulnerabilities, including—

    - a description of the process for identifying and incorporating key contributors to plant risk and risk insights into the design or construction

    - examples of design changes made based on risk information and insights

- if the PRA and alternative risk evaluations are used to select or support the selection of licensing basis events—

    - a description of the process for using risk information and insights for selecting licensing basis events, including the justification of metrics (e.g., event sequence frequency) and thresholds (e.g., separation between design-basis accidents and beyond-design-basis events)

    - a description of the treatment of uncertainty in the PRA and alternative risk evaluations in the process for selection of licensing basis events

    - a summary of the results from the use of the PRA and alternative risk evaluations for selection of licensing basis events

Self-Assessment and Peer Review

A PRA self-assessment is an acceptable tool for assessing the technical adequacy of a PRA performed in support of a CP application. The staff should determine whether a PRA self-assessment was performed for the CP PRAs commensurate with the design readiness. It is recognized that certain PRA elements may not be applicable or met. If the applicant's justification fails to provide the staff with an appropriate level of confidence in the models, results, and insights, the staff should conduct an audit of the applicant's PRA against the technical elements described in RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," to determine the PRA technical adequacy.

The reviewer should ensure that the CP application includes the following information for the self-assessment:

- a description of the PRA self-assessment, including the PRA standard(s) and guidance used to perform the self-assessment

- a summary of any limitations identified by the self-assessment, arising from the level of maturity of design and operational details

The above information will aid the review of the technical acceptability of the CP application PRA and its use in support of the CP application, including risk insights and results. The staff may accept a peer review using the PRA standards and related industry guidance, as endorsed by the NRC in RG 1.200, performed voluntarily at the CP application stage. A peer review provides additional confidence in the results of the PRA.

Hazard-Specific Information

Each section identified by an italicized heading below describes the purpose and contents of the hazard analysis or technical element considered in PRA and alternative risk evaluations, followed by guidance to the reviewer on each topic. The reviewer should verify that the CP application includes discussions and descriptions identified below, commensurate with the identified uses of the risk insights from the PRA and alternative risk evaluations, and the level of design maturity in the CP application.

Plant Operating State (POS) Analysis

The plant operating state (POS) analysis identifies operating evolutions (e.g., full-power, low-power, and shutdown types of conditions) important to risk. Each condition in which plant parameters are stable and similar is defined as a distinct POS. The purpose of the POS analysis is to identify and evaluate the entire spectrum of plant responses to off-normal conditions with a potential to lead to core damage and large release. Each POS in the POS analysis includes applicable initiating events and accident sequences, establishes system success criteria, and quantifies accident sequence frequencies. The set of identified POSs encompasses the entire spectrum of operations.

The reviewer should confirm that the CP application includes the following information from the POS analysis, consistent with the maturity of the design:

- the range of plant parameters and the selected representative parameter value chosen for each POS, for example, for power level or decay heat level, including typical POS entry times after plant trip; average reactor coolant system temperatures, configuration (e.g., intact, vented, or modified by dams, seals, and open penetrations), pressures, and water levels; and containment status (e.g., de-inerted, intact, open)

- a description of mitigation equipment available or expected to be available for each POS

- descriptions of activities that may lead to changes in the above parameters used to define the POS (e.g., drain down, filling and venting, dilution, fuel movement, and cooldown), including reactor coolant system pressure capability, presence of temporary hatches or penetrations, or nozzle dams or loop isolation

- information regarding the screening and grouping of POSs to facilitate an efficient but realistic estimation of CDF and LRF

- if bounding assessments or qualitative evaluations are performed to address certain evolutions, identification of the spectrum of accident sequences with the potential to lead to core damage and large release

Full-Power Internal Events PRAs

The reviewer should determine whether a full-power internal events PRA has been developed for the CP application, commensurate with the design maturity. In the CP application, consistent with DC/COL-ISG-028, Capability Category I of an NRC endorsed PRA standard is acceptable for PRAs, including the internal events PRA.

Initiating Event Analysis

Initiating events include perturbations to the steady-state operation of the plant that challenge plant control and safety systems and failures of plant control and safety systems that may perturb the steady-state operation of the plant, which could lead to core damage, radioactivity release, or both. The initiating event analysis identifies and characterizes the events that both challenge normal plant operation during power or shutdown conditions and require successful mitigation by plant equipment and personnel to prevent core damage from occurring. Initiating events are grouped by similarity of system and plant responses (based on the success criteria).

The reviewer should confirm that the CP application includes the following information on initiating event analysis for the full-power internal events PRA:

- a description of the systematic approach used to develop a comprehensive list of potential initiating events

- an identification of guidance (e.g., RG 1.200), PRA standards (e.g., the endorsed Level 1/LERF PRA Standard for LWRs, the Advanced Light-Water Reactor PRA Standard), data sources (e.g., operating experience), and techniques used to develop the comprehensive list of initiating events (e.g., failure modes and effects analysis, master logic diagram)

- an identification of initiating events that are screened from inclusion in the PRA and technical basis for the screening

- a description of how the initiating events that are not screened are categorized into initiating event categories or groups according to plant response and mitigation equipment

- a description of each initiating event

Accident Sequence Analysis

The objective of the accident sequence analysis is to model chronologically the possible accident progressions that can occur, starting from the initiating event modeled in the CP application PRA to its end state (e.g., successful mitigation, core damage, large release). The accident sequences account for the systems that are designed (and available) to mitigate the initiator based on defined success criteria. The event sequences also account for any operator actions performed to mitigate the accident based on the defined success criteria, plant operating procedures (e.g., plant emergency and abnormal operating procedures), and training.

The reviewer should confirm that the CP application includes the following information on the accident sequence analysis for the full-power internal events PRA:

- a summary of the event tree for each initiating event identified in the initiating event analysis, including a discussion of the sequences for each event tree

- a description of the (safety-related and non-safety-related) equipment reasonably expected to be used to mitigate initiators

- a description of plant-specific functional, phenomenological, and operational dependencies that impact significant event sequences in the event sequence structure

- a description of individual function mission times for each safety function and time windows for each operator action included in the PRA

Success Criteria Development

For an initiating event, success criteria identify the minimum system design and functional requirements to prevent or mitigate an undesirable end state. Success criteria are based on acceptable engineering analyses that represent the design and operation of the plant under consideration. For a safety function to be successful, the criteria depend on the initiator and the conditions created by the initiator.

The reviewer should confirm that the CP application includes the following information on success criteria for the full-power internal events PRA:

- the definitions of success criteria and mission time

- a summary of engineering analyses representing the available design and operation information performed to identify the success criteria

- descriptions of the success criteria for each initiating event or initiating event group, including the list of performance requirements (e.g., number of trains credited) and operator actions credited in the determination of success criteria

- identification of any computer code(s) used for analysis of success criteria, addressing the applicability of the code for evaluation of phenomena of interest

Systems Analysis

The objective of the systems analysis is to identify combinations of failures that can prevent a system from performing one of its safety functions. The systems analysis model includes failures of system hardware and instrumentation and human failure events (HFEs). Modeling these failures accounts for dependencies among the frontline and support systems and distinguishes the specific equipment or human events that have a major impact on the system's ability to perform its function.

The reviewer should confirm that the CP application includes the following information on the systems analysis for the full-power internal events PRA:

- descriptions of intra-system and inter-system dependencies and the methodology used for modeling common-cause failures, treatment of testing, and maintenance in the model

- identification of those passive safety systems that perform a safety function for any sequence

Passive Safety System Reliability

This section applies only to designs using passive systems for emergency core cooling or decay heat removal.

Passive safety systems rely on natural forces, such as gravity, to perform their safety functions. Such driving forces are small compared to those of pumped systems, and the uncertainty in their values, as predicted by a best-estimate thermal hydraulics analysis, can be of comparable magnitude to the predicted values themselves. Therefore, some accident sequences with a frequency high enough to impact results but not predicted to lead to core damage by a best estimate thermal hydraulics analysis may be predicted to lead to core damage when PRA models consider thermal hydraulic uncertainties for passive systems. Different approaches have been used to address this topic, including the use of a response surface approach based on sensitivity studies using the thermal-hydraulics code selected for success criteria analysis. Examples of approaches are those used for "Success Criteria and Thermal-Hydraulic Uncertainty ([Resolution of] Open Item 19.1.10.1-5)," Section 19.1.10.5 of "Chapter 19, Severe Accidents, AP1000 Final Safety Evaluation Report (FSER)," dated September 13, 2004 (Ref. 25); "Success Criteria and Passive System Uncertainty," Section 19.1.2.3.1 of NUREG-1966, "Final Safety Evaluation Report: Related to Certification of the Economic Simplified Boiling-Water Reactor Standard Design, Volume 4, Chapters 16-24", issued April 2014 (Ref. 26); and NuScale Power, LLC, Design Certification Application, "Passive System Uncertainty," Section 19.1.4.4.3, "FSER Chapter 19 Probabilistic Risk Assessment and Severe Accident Evaluation," dated July 23, 2020 (Ref. 27).

The reviewer should confirm that the CP application includes the following information on passive safety system reliability for the full-power internal events PRA:

- identification of all key thermal-hydraulics parameters that could affect the reliability of a passive system and introduce uncertainty into the determination of success criteria

- a description of how the key thermal-hydraulics phenomena are modeled as a failure mode

- if a thermal-hydraulics uncertainty analysis is performed—

    - a summary of its results and key insights

    - a discussion of the applicability of the thermal-hydraulics code used for the assessment

- if a thermal-hydraulics uncertainty analysis is not performed, a description of the plan to perform these analyses and reflect the insights into the design

Human Reliability Analysis

The objective of the human reliability analysis is to identify and define the HFEs that can negatively impact normal or emergency plant operation and quantify their probabilities. The HFEs associated with normal plant operation include the events that leave the system (as defined by the success criteria) in an unavailable state. The HFEs associated with emergency plant operation represent those human actions that, if not performed or performed incorrectly, do not allow the needed system to function. Only human errors of omission are considered in the scope of the systems analysis; errors of commission and malevolent acts are not considered in the scope of the systems analysis.

The reviewer should confirm that the CP application includes the following information on human reliability analysis for the full-power internal events PRA:

- an identification and description of HFEs that result in initiating events

- an identification and description of pre-accident and post-accident HFEs that impact the mitigation of initiating events

- an identification and treatment of dependent HFEs, including the basis for the lower bound of the joint human error probability used in the PRA

- any recovery action credit taken, including the justification for such credit

Data Analysis

The objective of the data analysis is to define the values of parameters for each basic event such that the PRA results provide realistic risk insights for the design. Data analysis includes the assignment of generic, design-specific, and plant-specific parameter value estimates (as applicable). Data analysis should account for SSC boundaries, failure modes, failure rates, and common-cause failures.

The reviewer should confirm that the CP application includes the following information on data analysis for the full-power internal events PRA:

- a discussion of sources of frequency and failure rates, with design-specific justification for use of generic estimates
- a design-specific justification for the failure rates used for first-of-a-kind components
- for safety features or components that require research and development (e.g., related to the failure rate used in the PRA), a description of the research and development program that will be conducted to resolve such issues at the OL stage

Level 2 Analysis

The reviewer should confirm that the CP application includes the following information on Level 2 analysis for the full-power internal events PRA:

- a description of the Level 2 PRA development, commensurate with the design in the CP application, including:
    - the grouping of Level 1 PRA core damage sequences
    - event trees and key phenomena for Level 2 PRA
    - the basis for excluding any severe accident phenomena
- a demonstration that the design at CP application conforms to the Commission's recommendations for containment performance for new reactors

Quantification

The reviewer should confirm that the CP application includes the following information on quantification for the full-power internal events PRA:

- estimates of CDF and LRF
- a list, with a summary description, of dominant sequences for CDF and LRF
- a list of dominant SSCs based on importance measures (e.g., Fussell-Vesely importance, risk achievement worth)
- a demonstration that the design conforms to the Commission's safety goals for new reactors

Uncertainty Analysis

The reviewer should confirm that the CP application includes the following information on the uncertainty analysis for the full-power internal events PRA:

- a summary of parametric uncertainty analysis performed with results, including the mean, 5th, and 95th percentile values for the CDF and LRF.
- a description of the process for identifying and dispositioning PRA model uncertainties for all the topic areas listed above, including identification of relevant guidance (e.g., RG 1.200, NUREG-1855, Revision 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making,” (Ref 29)).
- a list of sensitivity analyses performed, including, for each sensitivity, the uncertainty being addressed, the change in base parameter, and the results.
- a list of key assumptions and sources of uncertainty, including design features and design assumptions impacting the application and the stated uses of the PRA

Internal Flood

An internal flood PRA or an alternative risk evaluation of the risk from internal floods is acceptable for a CP application.

If an alternative risk evaluation is performed for internal floods for a CP application, the reviewer should confirm that the CP application includes the following information:

- a discussion of the alternative risk evaluation approach, including the systematic identification of potential internal flood initiating events
- a description of any screening analysis performed for any flood sources (initiators), including identification of design features (e.g., flood doors, berms, SSC elevations) relied on for screening the identified initiating events from inclusion in the nonalternative risk evaluation
- a description of the risk insights, including, as applicable, failures of SSCs and their consequences due to the internal flood initiators that were not screened
- identification of the key assumptions used in the evaluation
- a summary of any limitations associated with the internal flood assessment arising from the level of maturity of design and operational details

The internal flood PRA initiating events typically rely on the corresponding internal events PRA initiating events with modifications to include the impact of the identified flood scenarios. Flooding may cause initiating events and cause the failure of equipment used to respond to initiating events.

If an internal flood PRA is performed for a CP application, the reviewer should confirm that the CP application includes the following information:

- a summary of changes made to the internal events PRA to develop the internal flood PRA, addressing each of the internal events PRA elements listed in Table 1
- a description of the process for flood area partitioning, flood source analysis, and flood scenario analysis
- a description of any screening analysis performed for any flood sources (initiators), including identification of design features (e.g., flood doors, berms, SSC elevations) relied on for screening the identified initiating events from inclusion in the internal flood PRA
- a summary of any limitations associated with the internal flood PRA arising from the level of maturity of design and operational details

Internal Fire

An internal fire PRA or an alternative risk evaluation of the risk from internal fire is acceptable for a CP application.

If an alternative risk evaluation for internal fires is performed for a CP application, the reviewer should confirm that the CP application includes the following information:

- a discussion of the alternative risk evaluation approach, including the systematic identification of potential internal fire-initiating events
- a description of any screening analysis performed for any fire sources (initiators), including identification of any design features (e.g., physical separation, fire barriers, dampers) relied on for screening the identified initiating events from inclusion in the alternative risk evaluation
- a description of the risk insights, including, as applicable, failures of SSCs and the consequences of those failures due to the internal fire initiators that were not screened
- a discussion of any alternative shutdown locations and corresponding capabilities
- identification of the key assumptions used in the evaluation
- a summary of any limitations associated with the internal fire evaluation arising from the level of maturity of design and operational details (e.g., cable routing)

If an internal fire PRA is performed for a CP application, the reviewer should confirm that the CP application includes the following information:

- a summary of changes made to the internal events PRA to develop the internal fire PRA, addressing each of the internal events PRA elements listed in Table 1

- a description of the process for fire area partitioning, fire source analysis, and fire scenario analysis (including the control room and alternate shutdown locations)

- a description of any screening analysis performed for any fire sources (initiators), including identification of any design features (e.g., physical separation, fire barriers, dampers) relied on for screening the identified initiating events from inclusion in the internal fire PRA

- a summary of any limitations associated with the internal fire PRA arising from the level of maturity of design and operational details (e.g., cable routing)

Seismic

An alternative risk evaluation (i.e., a PRA-based seismic margins assessment (SMA)) or a seismic PRA may be used to support an LWR CP application.

For a PRA-based SMA, a design response spectra (DRS) representative of multiple sites may be used. Both the design and site-specific earthquake ground motion must satisfy 10 CFR Part 50, Appendix S and 10 CFR Part 100, "Reactor Site Criteria." The spectra are characterized by horizontal and vertical response spectra. If a PRA-based SMA is performed for a CP application, the reviewer should confirm that the CP application includes the following information:

(1) Seismic Hazard Input

- for an applicant using site-specific response spectra:

    - a depiction of the ground motion response spectra (GMRS) or site-specific safe-shutdown earthquake (SSE)

    - a depiction of the review-level earthquake (RLE)–1.67 times the GMRS or site-specific SSE

- for an applicant using DRS representative of an envelope of multiple sites—

    - a demonstration that the site-specific GMRS or SSE characterized by horizontal and vertical response spectra is bounded by the DRS

    - a depiction of an RLE–1.67 times the DRS defined as the SSE

- identification of any site-specific seismic-induced initiating events (e.g., slope stability, liquefaction, dam failure), including discussion of the approach

(2) Seismic Fragility Evaluation

- summary description of the systematic process used to develop the seismic equipment list (SEL)

- identification of seismically induced failures of SSCs that are not explicitly modeled in the internal events PRA and structural failures that could cause widespread equipment failures

- summary of seismic correlation assumptions

- list of the key SSC fragility parameters (e.g., high confidence of low probability of failure (HCLPF) values, median capacities, and logarithmic standard deviation of the fragilities for the SSCs on the SEL), including—

  - description of the method(s) used for the derivation of SSC fragilities, including a summary of how the failure probability is related to the ground motion parameter

  - identification of sources of information and justification for applicability of generic fragilities

(3) Systems and Accident Sequence Analysis

- a summary of the process for identifying site-specific seismic-induced initiating events, including the operating modes, event trees, fault trees, and accident sequences considered in the analysis with a basis for their selection

- a description of the development of the PRA-based SMA, including—

  - changes made to the internal events PRA model

  - modeling of passive components (e.g., tanks, heat exchangers, piping) and structural failures, including containment, and correlated failures

  - modeling of random failures and human actions specific to the PRA-based SMA and changes to the modeling of human actions to account for seismic events

- a description of failures that are assumed to lead directly to core damage or large release

(4) Sequence-Level and Plant-Level HCLPF Assessment

- a description of the calculated sequence-level and plant-level HCLPF capacities for the operating modes considered, including—

  - a discussion of the method used to calculate sequence-and plant-level HCLPF capacities (e.g., MIN-MAX)

  - identification of the SSCs that limit the plant-level HCLPF capacity

- identification of key assumptions and sources of uncertainty that can impact insights and results, including those arising from level of design maturity at the CP stage that lack as-built and as-operated details

- a description of the process for tracking assumptions and sources of uncertainty

- identification of any scenarios in which combinations of seismic failures, random events, and failures of human actions could result in an effective seismic capacity less than the RLE

- key results and insights, such as—

    - dominant seismically induced initiating events

    - dominant sequences

    - dominant functions, SSCs, and operator actions

    - identification of any potential vulnerabilities in the design

If a seismic PRA is performed to support an LWR CP application, the reviewer should confirm that the CP application includes the following information:

(1) Seismic Hazard Input

- a description of probabilistic seismic hazard analysis performed to develop the site-specific seismic hazard curves and any changes to the seismic hazard curves used in the seismic PRA

- a depiction of the site-specific response spectra with the technical basis for their development

- identification of site-specific seismic-induced initiating events (e.g., slope stability, liquefaction, dam failure), including discussion of the approach

(2) Seismic Fragility Evaluation

- a summary description of the systematic process used to develop the SEL

- identification of seismically induced failures of SSCs that are not explicitly modeled in the internal events PRA and structural failures that could cause widespread equipment failures

- a summary of seismic correlation assumptions

- a list of the key SSC fragility parameters (e.g., HCLPF values, median capacities, and logarithmic standard deviation of the fragilities for the SSCs on the SEL), including—

    - a description of the method(s) used for the derivation of the design-specific SSC fragilities, including a summary of how the SSC failure probability is related to the ground motion parameter

    - identification of sources of information and justification for applicability for the generic fragilities used

(3) Plant Systems Analysis

- a summary of the operating modes, accident sequences, event/fault trees, and damage levels considered in the analysis with a basis for their selection

- a description of the development of the seismic PRA, including—

    - changes made to the internal events PRA model

    - modeling of passive components (e.g., tanks, heat exchangers, piping) and structural failures, including containment, and correlated failures

    - modeling of random failures and human actions specific to the seismic PRA and changes to the modeling of human actions to account for seismic events

- a description of failures that are assumed to lead directly to core damage or a large release

- key results and insights such as—

    - the plant-level HCLPF

    - identification of any scenarios where combinations of seismic failures, random events, and failures of human actions could result in an effective seismic capacity less than the RLE

    - dominant seismically induced initiating events

    - dominant sequences and cutsets

    - dominant functions, SSCs, and operator actions

    - identification of any potential vulnerabilities in the design

- a description of the assumptions and sources of uncertainty for hazard, fragility, and plant response that can impact insights and results, including those arising from—

    - level of design maturity at CP application

  - lack of as-built and as-operated details

- identification of any sensitivity analyses performed to account for assumptions and sources of uncertainty

- a list of key assumptions and sources of uncertainty, including design features and design assumptions impacting the application and the stated uses of the seismic PRA

Nonseismic Hazards

A key feature of a PRA is that a wide spectrum of potential hazards in terms of magnitude and frequency of occurrence are systematically surveyed to ensure that significant contributors to plant risk are not inadvertently excluded. Table D-1 in Appendix D to RG 1.200 lists additional hazards that a reviewer should consider in evaluating an application. Non-seismic hazards may include additional hazards not listed in Table D-1. Non-seismic hazards may be evaluated using hazards screening, if applicable, conservative estimates of risk, or a non-seismic hazards PRA to support an LWR CP application.

Hazards Screening

The objective of the non-seismic hazards screening analysis is to adequately justify exclusion of a hazard or hazard group from the PRA model or alternative risk assessment.

If the applicant performs screening for any non-seismic hazard, including the hazards listed in table D-1 of Appendix D to RG 1.200, the reviewer should confirm that the CP application includes the following information for each hazard that screens out:

- a discussion of the basis for site-specific screening, identifying, if applicable, the corresponding criteria in the PRA standard endorsed in RG 1.200

- a description of the hazard screening analysis, including the applicability of data used for occurrence frequency in the analysis for the CP site

- identification of assumptions and sources of uncertainty for the screening of each screened hazard, including key assumptions that can impact the results of the screening

- identification of SSCs and design features credited in and necessary for screening of each screened hazard

Conservative Estimate of Risk from Non-seismic Hazards Using Alternative Risk Evaluations

If an applicant cannot screen out a non-seismic hazard based on a qualitative evaluation or quantitative screening analysis, the applicant may perform a conservative assessment of risk and demonstrate that the CP site is within the bounds of the parameters used for the conservative assessment. If the applicant performs a conservative analysis for any non-seismic hazard, including the hazards listed in table D-1 of appendix D to RG 1.200, the reviewer should confirm that, for each of these hazards, the CP application includes the following information:

(1) Hazard Input

- a description of the hazard frequency of occurrence at different intensities of the hazard for the CP site using a site-specific probabilistic evaluation

- a description of the historical data or a phenomenological model, or a mixture of the two, that is used for the hazard frequency development

(2) Fragility Evaluation

- a description of the systematic process used to develop the hazard safe-shutdown equipment list (SSEL)

- identification of hazard-induced failures of SSCs that are not explicitly modeled in the internal events PRA and structural failures that could cause widespread equipment failures

- correlation assumptions

- a description of the systematic process and assumptions used to determine the governing failure mode(s) for the SSCs on the SSEL

- a summary of the key SSC fragilities, including—

    - a description of the methods used for derivation if design-specific fragilities are used

    - identification of sources of information and justification for applicability if generic fragilities are used

(3) Plant Systems Analysis

- a summary of the operating modes, accident sequences, and event or fault trees, and the damage levels considered in the analysis with a basis for their selection

- a description of the development of the other hazards PRA, including changes made to the internal events PRA model, modeling of passive components, structural failures, correlated failures, random failures, and human actions

- a description of significant failures that can lead to core damage and large release

- key results and insights, including risk-significant SSCs, dominant cutsets, and dominant sequences

- a list of analysis assumptions and sources of uncertainty for hazard, fragility, and plant response that can impact insights and results, including those arising from—
    - o level of design maturity at the CP stage
    - o lack of as-built and as-operated details
- any sensitivity analyses performed to address assumptions and sources of uncertainty

Nonseismic Hazard PRA

A PRA for non-seismic hazards, including those identified in table D-1 of Appendix D to RG 1.200, may be used to quantify risk if the hazard is not screened out by either qualitative screening evaluation or a quantitative screening analysis, and a conservative analysis is not performed. For any such hazard, the reviewer should confirm that the CP application includes the following information:

(1) Hazard Input

- a description of the hazard frequency of occurrence at different intensities of the hazard for the CP site (including the most severe events reported for the site and surrounding area) using a site-specific probabilistic evaluation
- a description of the historical data or a phenomenological model, or a mixture of the two used for the hazard frequency development

(2) Fragility Evaluation

- a description of the systematic process used to develop the hazard SSEL
- identification of hazard-induced failures of SSCs that are not explicitly modeled in the internal events PRA and structural failures that could cause widespread equipment failures
- correlation assumptions
- a description of the systematic process and assumptions used to determine the governing failure mode(s) for the SSCs on the SSEL
- a summary of the key SSC fragilities, including—
    - o a description of the methods used for derivation if design-specific fragilities are used
    - o identification of sources of information and justification for applicability if generic fragilities are used

(3) Plant Systems Analysis

- a summary of the operating modes, accident sequences, and event or fault trees, and damage levels considered in the analysis with a basis for their selection

- a description of the development of the other hazards PRA, including changes made to the internal events PRA model, modeling of passive components, structural failures, correlated failures, random failures, and human actions

- a description of significant failures that can lead to core damage and large release

- key results and insights, including risk-significant SSCs, dominant cutsets, and dominant sequences

- a list of analysis assumptions and sources of uncertainty for hazard, fragility, and plant response that can impact insights and results, including those arising from

  - the level of design maturity at CP stage

  - the lack of as-built and as-operated details

- any sensitivity analyses performed to address assumptions and sources of uncertainty

Low Power and Shutdown

A low-power and shutdown (LPSD) PRA or an alternative risk evaluation of the risk from LPSD operations is acceptable for a CP application.

If an alternative risk evaluation of LPSD operations is performed for a CP application, the reviewer should confirm that the following information is included:

- a discussion of the alternative risk evaluation approach, including the systematic identification of potential LPSD initiating events based on the submitted POS analysis

- a description of any analysis performed to screen POSs from inclusion in the LPSD PRA, including identification of any design features relied on for the screening

- a description of the LPSD risk insights (e.g., design features that minimize operator actions relied upon to mitigate shutdown initiating events) derived from the assessment

- identification of key assumptions used in the evaluation

- a summary of any limitations arising from the level of maturity of design and operational details

If an LPSD PRA is performed for a CP application, the reviewer should confirm that the following information is included:

- a summary of changes made to the internal events PRA to develop the LPSD PRA addressing each of the internal events PRA elements listed in Table 1 and demonstrating consistency with identified POSs

- a description, with justification, of any analysis performed to screen any POS from inclusion in the LPSD PRA, including identification of any design features relied on for the screening

- a summary of any limitations associated with the LPSD PRA arising from the level of maturity of design and operational details

PRA Development and Configuration Plan

The PRA configuration control program is based on available operational, maintenance, and procedural information. The PRA configuration control program addresses design-specific, site-specific, and plant-specific characteristics and evaluations of changes made to them. The reviewer should confirm that the CP application contains the following information:

- identification of PRA elements from RG 1.200 that are not met or not applicable, an explanation for the reason each identified element is not met or does not apply (e.g., lack of design maturity), and a description of the applicant’s plan for addressing the PRA elements identified as inapplicable or not met in the OL PRA

- the guidance and standards used to develop the PRA, including any commitments to the standards (and, if applicable, the capability categories) that will be met for the PRA supporting the OL application

- a description of the process to track assumptions and monitor inputs for PRA and alternative risk evaluations supporting the CP application

- a description of how new information will be collected and included in the PRA to maintain the PRA consistent with the as-built, as-to-be-operated plant design

- a description of how configuration control of computer models and codes used to support PRA inputs and quantification will be performed

- a description of how reviews of the PRA will be conducted (i.e., self-assessment, peer review, etc.), including the frequency of such reviews

- a description of when the PRA is to be updated or upgraded

Severe Accidents

In accordance with the Commission’s Severe Accident Policy Statement, the reviewer should determine whether the application considered a range of alternatives to reduce risk from severe accidents. The reviewer should evaluate the CP application's assessment of severe accident risk from events such as core-concrete interaction, steam explosion, high-pressure core-melt ejection, hydrogen combustion, and containment bypass. The reviewer should determine whether a severe accident, such as those listed above, is relevant to the design under review. The reviewer should evaluate whether the PRA and alternative risk evaluations consider severe accident vulnerabilities and address the prevention and mitigation of severe accidents. The reviewer should confirm that the CP application includes the following information on severe accidents:

- a description and analysis of design features for the prevention and mitigation of severe accidents, including an evaluation of the severe accident phenomena to assess their design relative to the containment performance goals as approved by SRM-SECY-93-087

- documentation of how the search for severe accident vulnerabilities is conducted, justification that the approach used to conduct the search is adequate, and the results of the search for severe accident vulnerabilities

- a description of how the overarching goal of the identification of severe accident vulnerabilities, which is to prevent the existence of an unacceptable likelihood or consequence of a severe accident, is achieved

- a description of improvements to plant design, operations, or maintenance that prevent or reduce the possibility, likelihood, or consequence of the identified severe accident

- a description of the analysis that has been performed for the CP application or will be performed as part of the OL application for each severe accident to understand the sequence and timing of events, phenomena, and how operators and other staff interact with and participate in the event sequence

<u>Regulatory Treatment of Nonsafety Systems for Designs with Passive Safety Systems</u>

The regulatory treatment of non-safety systems (RTNSS) process applies to designs with passive safety systems. More specifically, it applies to those non-safety-related SSCs that perform risk-significant functions and, therefore, are candidates for regulatory oversight. SECY-94-084 and SECY-95-132 describe the scope, criteria, and specific steps of the RTNSS process. SRP Section 19.3, "Regulatory Treatment of Nonsafety Systems for Passive Advance Light Water Reactors," contains corresponding review guidance. The RTNSS process applies broadly to those non-safety-related SSCS that perform risk significant functions and, therefore, are candidates for regulatory oversight. Systems included in the RTNSS program based on PRA insights and results include:

- Non-safety-related design features or functional capabilities with mitigation capability necessary to reduce the CDF or LRF below the Commission goals when credited in the PRA (RTNSS C).

- Non-safety-related SSCs whose failure results in PRA initiating events that result in passive safety system actuation and significantly affect CDF and LRF (RTNSS C).

- Non-safety-related SSCs relied upon to compensate for potential uncertainties associated with assumptions made in the PRA regarding passive systems and in the modeling of severe accident phenomenology (unless a reasonable justification is given for not doing so) (RTNSS C).

- Non-safety-related SSCs credited in meeting the Commission's containment performance goals (RTNSS D).

The reviewer should confirm that the CP application includes the following information on RTNSS:

- a description of the non-safety-related SSCs subject to RTNSS and their specified functions, including the specific RTNSS criteria that are met by the SSCs

- a discussion of how candidate risk significance is determined from the PRA, including numeric thresholds and their bases

- if active systems are determined to be risk-significant,[5] a description of the administrative controls on availability, or technical specifications and limiting conditions for operation

- a description of the augmented design standards that must be met by SSCs in the scope of the RTNSS program and standards for assuring that SSC functions will be achieved

- the regulatory treatment proposed for SSCs in the scope of the RTNSS program

IMPLEMENTATION

The staff will use the information discussed in this ISG to supplement the guidance in the SRP and in DNRL-ISG-2022-01 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, FORWARD FITTING, AND ISSUE FINALITY 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

5 One endorsed definition of "risk-significant" is found in RG 1.200, which defines it in general terms with reference to the definitions for "significant accident sequence" and "significant basic event/contributor" with quantitative bands. The NRC staff will review design-specific definitions of "risk-significant" and their justifications on a case-by-case basis.

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

Discussion to be provided in final ISG.

FINAL RESOLUTION

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

APPENDIXES

A. Resolution of Public Comments
B. References

Appendix A
Resolution of Public Comments

A notice of opportunity for public comment on this interim staff guidance (ISG) was published in the *Federal Register* (*insert FR Citation #)* on [date] for a 30–60 day comment period. [Insert number of commenters] provided comments which were considered before this ISG is issued in its final form.

Comments on this ISG are available electronically at the U.S. Nuclear Regulatory Commission's (NRC's) electronic reading room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can access ADAMS, which provides text and image files of the NRC's public documents. Comments were received from the following individuals or groups:

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

The comments and the staff responses are provided below.

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

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

- If the NRC staff agrees, explain why and provide a clear statement as to how the relevant language was revised or supplemented to address the comment. Include the following language at the end of the comment response: "The final ISG was changed by *<describe the change; if necessary, by quoting the newly revised language>*."

- If the NRC disagrees with a comment and no change was made to the generic communication, then explain why and provide the following language at the end of the comment response: "No change was made to the final ISG as a result of this comment."

Appendix B
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