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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

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

DRA-ISG-2026-01

PURPOSE

This interim staff guidance (ISG) clarifies the scope and depth of the staff's review of the description of risk assessment and severe accident information in the preliminary safety analysis report (PSAR) for a light-water power reactor construction permit (CP) application that uses risk assessment and severe accident information.

The guidance covers probabilistic risk assessment (PRA) and alternative risk evaluations. It supplements the guidance in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP) (Ref. 1).[1]

The review of a CP application for an LWR falls within the two-step licensing process under 10 CFR Part 50. The regulations in 10 CFR Part 50 do not require development of a PRA for a CP application, as reiterated by the Commission in Staff Requirements Memorandum (SRM)-SECY-22-0052, "Staff Requirements—SECY-22-0052—Proposed Rule: Alignment of Licensing Processes and Lessons Learned from New Reactor Licensing (RIN 3150-AI66)," dated November 20, 2024 (Ref. 4). This ISG provides staff review guidance for cases in which a CP applicant uses risk assessment and severe accident information to support its application.

BACKGROUND

The two-step licensing process under 10 CFR Part 50 involves the issuance of a CP based on preliminary design information and documented in a PSAR that allows an applicant to begin construction. When the design is essentially complete, the licensee will supply a final safety analysis report (FSAR) with the application for an operating license (OL). The FSAR describes the complete and final design of the facility as constructed; identifies the changes from the criteria, design, and bases in the PSAR; and discusses 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, using the guidance in the SRP. The NRC provided guidance to applicants for preparing COL applications in Regulatory Guide (RG) 1.206, Revision 0, "Combined License Applications for Nuclear Power Plants (LWR Edition)," issued

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[1] The SRP contains review guidance for an application to build and operate a light-water reactor (LWR), whether the application is submitted under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities" (Ref. 2), or 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants" (Ref. 3).

June 2007 (Ref. 5). The NRC has periodically updated some of the SRP guidance, and it issued RG 1.206, Revision 1, “Applications for Nuclear Power Plants,” in October 2018 (Ref. 6).

RG 1.70, Revision 3, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants: LWR Edition,” issued November 1978 (Ref. 7), offers some insights on the level of detail acceptable for PSARs in CP applications. However, this guidance has not been updated since 1978, and the insights may be limited to the degree that they do not account for subsequent requirements, NRC technical positions, novel design approaches, or advances in technical knowledge.

On October 31, 2022, the NRC staff issued interim staff guidance (ISG) DNRL-ISG-2022-01, “Safety Review of Light-Water Power Reactor Construction Permit Applications” (Ref. 8), to facilitate safety reviews of LWR CP applications and 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. It also provides guidance on the staff’s review of preliminary design information in the PSAR, including the description and safety assessment of the site where the facility is to be located.

DNRL-ISG-2022-01 does not provide specific information relevant to the review of PRA, alternative risk evaluations, or severe accident information supporting an LWR CP application, but it 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 ISG to clarify the scope and depth of the staff’s review of the description of risk assessment and severe accident information in the PSAR for a light-water power reactor CP application that uses risk assessment and severe accident information.

The staff has engaged with stakeholders in several public meetings on this topic. It considered the stakeholder views stated in these meetings in formulating the positions presented in this ISG.[2] This ISG replaces the draft white paper dated November 29, 2023 (Ref. 9), developed by the NRC staff, that was not issued as an official agency position on this subject and is available only as historical background information. The staff issued the draft ISG for public comment on January 16, 2025. The comment period lasted until February 18, 2025. The two sets of public comments and the NRC staff dispositions are provided in Appendix A.

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 include 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

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

PRA, can help demonstrate that the application complies with the regulations and follows Commission policy, including the following:

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

6. supporting the adequacy of the plant's defense-in-depth capability

The PRA and alternative risk evaluations may also be used for other purposes, such as a part of or a basis for the determination of licensing-basis events.

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] then 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 obtained from the CP PRA will need to aid the development of a PRA to support an OL application, including the confirmation of changes made during construction from 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.

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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). Table 2 lists examples of alternative risk evaluations that may be considered for a CP application.

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 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 PRAs and alternative risk evaluations in CP applications is to focus the NRC staff's review on those aspects of the design that contribute most to safety, minimizing attention to issues of low risk or low safety significance. Consistent with the NRC's use 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 that use risk assessment and severe accident information to support the application and that do not use the Licensing Modernization Project (LMP) framework.[4]

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 that the staff relies on 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 whether the applicant demonstrates conformance to relevant Commission policy (e.g., safety goals).

The guidance in this document considers the role of PRAs and alternative risk evaluations, the severe accident analysis practicable 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

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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. 11). With clarifications and points of emphasis, 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. 12), 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," issued August 2021 (Ref. 13), 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" (Ref. 14), with clarifications and additions.

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," issued November 2016 (Ref. 15), 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 (ASME)/American Nuclear Society (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" (PRA Standard), issued February 2009 (Ref. 16), that are not applicable or cannot be achieved as written for the design certification (DC) and COL application stages. The NRC 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 is needed for a high-level requirement.

Applicable Regulations, Commission Policy Statements, and Guidance Documents

The primary 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), a CP application for a stationary power reactor 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 FSAR 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 that 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. When 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 typically be consistent with the type and level of detail of information provided at the OL stage. In such cases, PRA acceptability should be generally consistent with that for a COL applicant, as discussed in SRP Chapter 19, "Severe Accidents," RG 1.200, and DC/COL-ISG-028, and is not discussed further in this guidance.

LWR CP applications may use PRAs and alternative risk evaluations to support meeting specific regulations, such as 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 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. 17)
- "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," dated August 8, 1985 (Ref. 18)
- "Safety Goals for the Operations of Nuclear Power Plants; Policy Statement; Correction and Republication," dated August 21, 1986 (Ref. 19)
- "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," dated August 16, 1995 (Ref. 20)
- SRM-SECY-90-016, "SECY-90-016—Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," dated June 26, 1990 (Ref. 21)
- SRM-SECY-93-087, "SECY-93-087—Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," dated July 21, 1993 (Ref. 22)
- SRM-SECY-94-084, "SECY-94-084—Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems," dated June 30, 1994 (Ref. 23)
- SRM-SECY-95-132, "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. 24)
- SRM-SECY-12-0081, "Staff Requirements—SECY-12-0081—Risk-Informed Regulatory Framework for New Reactors," dated October 22, 2012 (Ref. 25)

- 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. 26)

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

- RG 1.200, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities”

- With clarifications and qualifications, RG 1.200 endorses ASME/ANS RA-Sa–2009, “Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications,” Addendum A to RA-S–2008

- 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 November 10, 2016

- SRP chapter 19

Uses of Probabilistic Risk Assessment Information in a Construction Permit Application

Results and insights from PRAs and alternative risk evaluations comprise 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 does the following:

- 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, initiating events, and radiological sources 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 the limitations at the CP stage or resolving the 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:

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

- 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.

If 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 the qualitative and quantitative information presented in the PSAR, as well as the applicant's commitments.

Minimum Scope of Probabilistic Risk Assessment and Alternative Risk Evaluations for a Construction Permit Application

The staff should ensure that the applicant has evaluated all hazards for their impact on the risk from the design. If a PRA is used to support an LWR CP application under 10 CFR Part 50, a reviewer can refer to table 1 for a summary of the minimum scope of the PRA. However, the scope of the PRA may need to be greater, depending on the intended uses of PRA information for a given application. A reviewer can refer to table 2 for a summary of additional PRA and alternative risk evaluations that can be used to support a CP application.

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. The staff should consider whether any particular supporting requirement endorsed in industry standards may not be applicable, or cannot be achieved as written, for the CP stage and should consider the applicant's approach in determining the applicability of supporting requirements. The staff should review the applicant's justification that the scope and level of detail of any PRA or alternative risk evaluation are consistent with the intended uses of the information from those assessments to support the CP application. The staff should review the applicant's plan for assessing any risk contributors not addressed by a PRA or alternative risk evaluation.

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 (SMAs) and conservative assessments of non-seismic external hazards. The staff should confirm that these alternative risk evaluations incorporate site-specific information.

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 LERF at or before initial fuel load and to discontinue regulatory use of LRF and CCFP thereafter.

For 10 CFR Part 50 plants, the staff should consider whether the CP PRA uses LRF or LERF; the staff will ultimately determine whether the OL PRA uses LERF because an OL authorizes the loading of fuel consistent with the guidance in RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Ref. 27). 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.

Note that tables 1 and 2 are independent of each other; the staff can review a CP application with an internal events PRA by considering the minimum scope in table 1 while reviewing the remaining hazards using deterministic approaches. In addition, the staff can review a combination of the approaches presented in table 2 in regard to different hazards (e.g., seismic PRA and external flood alternative risk evaluation). Further, the staff should not apply table 2 if an applicant chooses to use a deterministic approach to address a particular hazard. Deterministic approaches, such as the design basis hazard levels discussed in RG 1.253, are outside the scope of this ISG.

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 a 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

* Table 2 is independent of table 1. Deterministic approaches, such as the design-basis hazard levels that may be used to address the hazards in table 2, are outside the scope of this ISG.

Submittal Information for the Probabilistic Risk Assessment and Alternative Risk Evaluations in a Construction Permit 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 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)

- the 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. The staff recognizes 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 to determine the PRA's technical adequacy. If the reviewer will need to rely on information identified during an audit to make the safety findings, the staff should ensure that information is available on the docket.

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 design maturity 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 standard(s) 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 the PRA and alternative risk evaluations, followed by guidance to the reviewer on each topic. The reviewer should verify that the CP application includes the discussions and descriptions identified below, commensurate with the identified uses of risk insights from the PRA and alternative risk evaluations and the level of design maturity in the CP application.

Plant Operating State Analysis

The plant operating state (POS) analysis identifies operating evolutions important to risk (e.g., full-power, low-power, and shutdown conditions). 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 spectrum of plant responses to off-normal conditions, with a potential to lead to core damage and large release. Each POS in the 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.

If the CP application includes a POS analysis, the reviewer should confirm that the CP application includes the following information from the 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 could lead to changes in the above parameters used to define the POS (e.g., draindown, 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, the identification of the spectrum of accident sequences with the potential to lead to core damage and large release

Full-Power Internal Events Probabilistic Risk Assessments

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 could 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 call for 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

- the identification of guidance (e.g., RG 1.200), PRA standards (e.g., the endorsed Level 1/LERF PRA Standard for LWRs), 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)

- the identification of initiating events that are screened from inclusion in the PRA and the 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 equipment (safety-related and non-safety-related) 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
- the identification of any computer codes used for the analysis of success criteria, addressing the applicability of the code for the 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- and inter-system dependencies and the methodology used for modeling common-cause failures, treatment of testing, and maintenance in the model

- the identification of 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 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 have an impact on results, but not predicted to lead to core damage by a best estimate thermal-hydraulics analysis, could 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 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 the following:

- Section 19.1.10.5, “Success Criteria and Thermal-Hydraulic Uncertainty ([Resolution of] Open Item 19.1.10.1-5),” of Chapter 19, “Severe Accidents,” of the AP1000 Final Safety Evaluation Report (FSER), dated September 13, 2004 (Ref. 28)

- Section 19.1.2.3.1, “Success Criteria and Passive System Uncertainty,” 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. 29)

- Section 19.1.4.4.3, “Passive System Uncertainty,” of Chapter 19, “Probabilistic Risk Assessment and Severe Accident Evaluation,” of the NuScale Power, LLC, DC FSER dated July 23, 2020 (Ref. 30)

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

- the 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 in an unavailable state (as defined by the success criteria). The HFEs associated with emergency plant operation represent those human actions that, if not performed or if performed incorrectly, do not allow the needed system to function. Only human errors of omission—not errors of commission or malevolent acts—are 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:

- identification and description of HFEs that result in initiating events
- identification and description of pre- and post-accident HFEs that impact the mitigation of initiating events
- 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 data analysis is to define the 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 following:
    - the grouping of Level 1 PRA core damage sequences

    - event trees and key phenomena for a 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 new reactor containment performance

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)

- an analysis of whether 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 percentile, and 95th percentile values for the CDF and LRF

- a description of the process for identifying and dispositioning PRA model uncertainties for all the topics listed above, including the identification of relevant guidance (e.g., RG 1.200 or NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," issued March 2017 (Ref. 31))

- 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 the identification of design features (e.g., flood doors, berms, SSC elevations) relied on to screen the identified initiating events from inclusion in the alternative 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

- the 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 the design and operational details

The initiating events for the internal flood PRA typically rely on the corresponding initiating events in the internal events PRA, with modifications to include the impact of the identified flood scenarios. Flooding can cause initiating events and 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 the 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 the 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 the 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, SSC failures 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

- the 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 the 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 the 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 the design and operational details (e.g., cable routing)

<u>*Seismic*</u>

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

For a PRA-based SMA, design-response spectra (DRS) representative of multiple sites may be used. The design and site-specific earthquake ground motion must both satisfy 10 CFR Part 50, Appendix S, “Earthquake Engineering Criteria for Nuclear Power Plants,” and 10 CFR Part 100, “Reactor Site Criteria” (Ref. 32). 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

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

- 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

(2) seismic fragility evaluation

- a summary description of the systematic process used to develop the seismic equipment list (SEL)
- the 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., 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—
  - a description of the method(s) used to derive SSC fragilities, including a summary of how the failure probability relates to the ground motion parameter
  - the identification of the 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, seismically 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- and plant-level HCLPF assessment

- a description of the calculated sequence- 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)

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

- the identification of key assumptions and sources of uncertainty that can impact insights and results, including those arising from the 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

- the 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
    - the 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

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

(2) seismic fragility evaluation

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

- the 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—
    - o a description of the method(s) used to derive the design-specific SSC fragilities, including a summary of how the SSC failure probability is related to the ground motion parameter
    - o the 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—
    - o changes made to the internal events PRA model
    - o modeling of passive components (e.g., tanks, heat exchangers, piping) and structural failures, including containment, and correlated failures
    - o 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—
    - o the plant-level HCLPF
    - o the 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
    - o dominant seismically induced initiating events
    - o dominant sequences and cutsets
    - o dominant functions, SSCs, and operator actions
    - o the identification of any potential vulnerabilities in the design
- a description of the assumptions and sources of uncertainty for hazard, fragility, and plant response that could impact insights and results, including those arising from the level of design maturity at CP application and lack of as-built and as-operated details
- the 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 stated uses of the seismic PRA

Non-seismic Hazards

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

Hazard Screening

The objective of non-seismic hazard screening analysis is to adequately justify the 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 is screened 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
- the 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
- the 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 the CP application includes the following information for each of these hazards:

(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)

- the 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—

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

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

(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 could 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—

    - level of design maturity at the CP stage
    - lack of as-built and as-operated details

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

<u>Non-seismic Hazard Probabilistic Risk Assessment</u>

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 if 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

- the 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

    - the 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, 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 could 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 could impact the insights and results, including those arising from—

    - the level of design maturity at the 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 may be used to support 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 the identification of any design features relied on for screening

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

- the identification of key assumptions used in the evaluation

- a summary of any limitations arising from the level of maturity of the 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 the identification of any design features relied on for screening

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

Probabilistic Risk Assessment Development and Configuration Plan

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

- the identification of PRA elements from RG 1.200 that are not met or are 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 not applicable 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), 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 severe accident phenomena to assess the design relative to the containment performance goals, as approved by SRM-SECY-93-087
- documentation of how the search for severe accident vulnerabilities was 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 identifying severe accident vulnerabilities (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, in order to understand

the sequence and timing of events, phenomena, and how operators and other staff interact with and participate in the event sequence

Regulatory Treatment of Nonsafety Systems for Designs with Passive Safety Systems

The regulatory treatment of nonsafety systems (RTNSS) process applies to designs with passive safety systems. More specifically, it applies to non-safety-related SSCs that perform risk-significant functions and are, therefore, candidates for regulatory oversight. SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” dated March 28, 1994 (Ref. 33), and SECY-95-132, “Policy and Treatment Issues Associated with RTNSS in Passive Plant Designs (SECY-94-084),” dated May 22, 1995 (Ref. 34), describe the scope, criteria, and specific steps of the RTNSS process. RTNSS SSCs may not be identified for a particular design following the RTNSS process. SRP Section 19.3, “Regulatory Treatment of Nonsafety Systems for Passive Advanced Light Water Reactors,” issued June 2014 (Ref. 35), contains corresponding review guidance. The RTNSS process applies broadly to non-safety-related SSCs that perform risk-significant functions and are, therefore, candidates for regulatory oversight.

The RTNSS program includes the following systems based on PRA insights and results:

- 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 cause passive safety system actuation and significantly affect CDF and LRF (RTNSS C)

- non-safety-related SSCs relied on 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)

If the CP application includes an RTNSS evaluation, the reviewer should confirm that the application includes the following information on RTNSS and is consistent with SRP section 19.3:

- 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 SSCs must meet in the scope of the RTNSS program, and standards for ensuring 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 CP applications. Issuance of this final ISG would not constitute backfitting, as defined in 10 CFR 50.109, "Backfitting" (the "Backfit Rule"), and as described in NRC Management Directive 8.4, "Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests," dated September 20, 2019 (Ref. 36); 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 based its position on 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 for the 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 to 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 10 CFR 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 refers to a license, such as an early site permit, or an NRC regulatory approval, such as a DC rule (or both), for which specified issue finality provisions apply. At present, the NRC staff does not intend to impose the positions represented in this ISG in a manner that constitutes backfitting or is inconsistent with any

[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.

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 that would allow the staff to impose the position.

- 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.

EXECUTIVE ORDER 12866, REGULATORY PLANNING AND REVIEW

Executive Order 12866, "Regulatory Planning and Review," dated September 30, 1993, provides that the Office of Information and Regulatory Affairs will determine whether a regulatory action is significant as defined by Executive Order 12866 and will review significant regulatory actions. The Office of Information and Regulatory Affairs determined that this final ISG is not a significant regulatory action under Executive Order 12866.

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.

REFERENCES

1. U.S. Nuclear Regulatory Commission (NRC), 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. *U.S. Code of Federal Regulations* (CFR), "Domestic Licensing of Production and Utilization Facilities," Part 50, Chapter I, Title 10, "Energy."

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

4. NRC, Staff Requirements Memorandum (SRM)-SECY-22-0052, "Staff Requirements—SECY-22-0052—Proposed Rule: Alignment of Licensing Processes and Lessons Learned From New Reactor Licensing (RIN 3150-A166)," November 20, 2024 (Agencywide Documents Access and Management System Accession No. ML24326A003).

5. NRC, Regulatory Guide (RG) 1.206, Revision 0, "Combined License Applications for Nuclear Power Plants," June 2007 (ML070720184).

6. NRC, RG 1.206, Revision 1, "Applications for Nuclear Power Plants," October 2018 (ML18131A181).

7. NRC, RG 1.70, Revision 3, "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," November 1978 (ML011340122).

8. NRC, DNRL-ISG-2022-01, "Safety Review of Light-Water Power Reactor Construction Permit Applications," October 31, 2022 (ML22189A099).

9. NRC, "Guidelines for Risk Assessment and Severe Accident Information in a Light-Water Reactor Construction Permit Application," November 29, 2023 (ML23326A185).

10. NRC, RG 1.200, "Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities," https://www.nrc.gov/reading-rm/doc-collections/reg-guides/power-reactors/rg/division-1/division-1-181.html.

11. NRC, 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," June 2020 (ML20091L698).

12. Nuclear Energy Institute (NEI) 18-04, Revision 1, "Risk-Informed Performance-Based Technology-Inclusive Guidance for Non-Light-Water Reactor Licensing Basis Development," August 2019 (ML19241A472).

13. 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," August 2021 (ML22060A190).

14. NRC, RG 1.253, "Guidance for a Technology-Inclusive Content Application Methodology to Inform the Licensing Basis and Content of Applications for Licenses, Certifications, and Approvals for Non-Light-Water Reactor," April 2024 (ML23269A047).

15. NRC, 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," November 2016 (ML16130A468).

16. American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) Standard 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," February 2009.

17. NRC, "Policy Statement on the Regulation of Advanced Reactors," *Federal Register*, Vol. 73, No. 199, p. 60612 (73 FR 60612), October 14, 2008.

18. NRC, "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants," *Federal Register*, Vol. 50, No. 153, p. 32138 (50 FR 32138), August 8, 1985.

19. NRC, "Safety Goals for the Operation of Nuclear Power Plants; Policy Statement; Correction and Republication," *Federal Register*, Vol. 51, No. 162, p. 30028 (51 FR 30028), August 21, 1986.

20. NRC, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," *Federal Register*, Vol. 60, No. 158, p. 42622 (60 FR 42622), August 16, 1995.

21. NRC, SRM-SECY-90-016, "SECY-90-016—Evolutionary Light Water Reactor (LWR) Certification Issues and Their Relationships to Current Regulatory Requirements," June 26, 1990 (ML003707885).

22. NRC, SRM-SECY-93-087, "SECY-93-087—Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," July 21, 1993 (ML003708056).

23. NRC, SRM-SECY-94-084, "SECY-94-084—Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems," June 30, 1994 (ML003708098).

24. NRC, SRM-SECY-95-132, "SECY-95-132—Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs," June 28, 1995 (ML003708019).

25. NRC, SRM-SECY-12-0081, "Staff Requirements—SECY-12-0081—Risk-Informed Regulatory Framework for New Reactors," October 22, 2012 (ML12296A158).

26. NRC, 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).

27. NRC, RG 1.174, Revision 3, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," January 2018 (ML17317A256).

28. NRC, Chapter 19, "Severe Accidents," AP1000 Final Safety Analysis Report, September 13, 2024 (ML033290638).

29. NRC, NUREG-1966, "Final Safety Evaluation Report: Related to Certification of the Economic Simplified Boiling-Water Reactor Standard Design," Volume 4, Chapters 16–24, April 2014 (ML14100A187).

30. NRC, Chapter 19, "Probabilistic Risk Assessment and Severe Accident Evaluation," NuScale Power, LLC, Design Certification Application Final Safety Evaluation Report, July 23, 2020 (ML20205L410).

31. NRC, NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," March 2017 (ML17062A466).

32. CFR, "Reactor Site Criteria," Part 100, Chapter I, Title 10, "Energy."

33. NRC, SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs,” March 28, 1994 (ML003708068).

34. NRC, SECY-95-132, “Policy and Treatment Issues Associated with RTNSS in Passive Plant Designs (SECY-94-084),” May 22, 1995 (ML003708005).

35. NRC, NUREG-0800, Section 19.3, Revision 0, “Regulatory Treatment of Nonsafety Systems for Passive Advanced Light Water Reactors,” June 2014 (ML14035A149).

36. NRC, Management Directive 8.4, “Management of Backfitting, Forward Fitting, Issue Finality, and Information Requests,” September 20, 2019 (ML18093B087).

APPENDIX A

Resolution of Public Comments on Draft Interim Staff Guidance DRA-ISG-2024-XX, "Content of Risk Assessment and Severe Accident Information in Light-Water Power Reactor Construction Permit Applications"

Comments on the draft interim staff guidance (ISG) are available electronically at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can access the Agencywide Documents Access and Management System (ADAMS), which provides text and image files of U.S. Nuclear Regulatory Commission (NRC) public documents.

The following table lists the comments the NRC received on the draft ISG.

Comment Number	ADAMS Accession No.	Commenter Affiliation	Commenter Name
NRC-2024-0217-DRAFT-0001	ML25063A219	Nuclear Energy Institute	V. Anderson
NRC-2024-0217-DRAFT-0002	ML25063A222	NuScale Power	Mark Shaver

The following table lists each public comment by number, as given in the table above. It provides the original comment as written by the commenter.

Comment Identifier	Topic	Specific Comment	NRC Staff Response
NRC-2024-0217-DRAFT-0001-1	General	The first sentence in the Guidance section of the ISG states that "This document provides guidance to the staff on the acceptability of the description of the [probabilistic risk assessment] PRA and its results and severe accident information in [preliminary safety analysis reports] PSARs." Please clarify if this is intended to mean guidance on acceptability of the descriptions of PRA information in the PSAR, or acceptability of the actual PRA model information and results provided in the PSAR.	The NRC staff agrees with the comment that additional clarity is needed to specify the guidance is intended for acceptability of the descriptions of PRA information in the PSAR. The ISG provides guidance on the acceptability of PRA information included in a construction permit (CP) application submittal if PRA information is used to support the submittal. The ISG describes the criteria the staff should use to review the information included in the PSAR. The third paragraph of the "Guidance" section in the ISG discusses the staff's review of the acceptability of the PRA in relation to Regulatory Guide (RG) 1.200 and DC/COL-ISG-028. The NRC staff also addresses this comment in the response to NRC-2024-0217-DRAFT-0001-2.
NRC-2024-0217-DRAFT-0001-2	General	The draft ISG provides guidance on the PRA information required for a construction permit application (CPA) submittal. However, for each of the items discussed, it is not consistently clear if the information is required to be included in the PSAR or if it should be available in separate source documents and analyses	The NRC staff agrees with the need to clarify the ISG insofar as it may appear to provide guidance on the content of a construction permit (CP) application. The purpose of the ISG is to provide review guidance to the staff. To the extent the text of the ISG appears to provide guidance to applicants on the content of CP applications, the staff has revised the ISG to clarify that the ISG is providing staff review guidance. As is always the case, if information is not submitted on the docket of the application, the staff cannot rely on that information as a basis for a decision on the

Comment Identifier	Topic	Specific Comment	NRC Staff Response
		supporting the CPA. This should be clarified in the ISG.	application. Accordingly, if an applicant seeks to rely on information as a basis for the acceptability of the application, the information should be included in the PSAR. Confirmatory or supporting information (not to be confused with "supporting requirements" designated in PRA standards) need not be included in the PSAR. The ISG provides guidance on the acceptability of PRA information submitted in a CP application. The ISG describes guidance for the staff review to determine the adequacy of the information included in the PSAR. The NRC staff updated the "Purpose" section of the final ISG to read: *This interim staff guidance (ISG) clarifies the scope and depth of the staff's review of the description of risk assessment and severe accident information in the preliminary safety analysis report (PSAR) for a light-water power reactor construction permit (CP) application that uses risk assessment and severe accident information.* The NRC staff updated the "Background" section of the final ISG to read: *As stated above, the staff has developed this ISG to clarify the scope and depth of the staff's review of the description of risk assessment and severe accident information in the PSAR for a light-water power reactor CP application that uses risk assessment and severe accident information.* The staff has made changes throughout the ISG to conform to these updates.
NRC-2024-0217-DRAFT-0001-3	General	The draft ISG provides minimum elements and scope for a CPA, some of which may not be available at the current stage of design and PRA model development. Please note/clarify which of these are required for acceptance of the CPA submittal for review, and which could be provided later as supplemental information during the review process. Please also clarify under which circumstances PRA information would be required to support a CPA, and which it would not be (e.g., Part 50 vs. Part 52).	The NRC staff agrees with the first part of the comment and disagrees with the second part of the comment. This ISG provides application review guidance to the staff and is not intended to provide acceptance and docketing guidance for an applicant or the staff. The risk information is to be commensurate with the application for which it is intended and the role the results play in the integrated decision-making process. Consistent with the discussion in the ISG section titled "Guidance," on determining the applicability of supporting requirements in industry standards to a CP application for a light-water reactor (LWR), the NRC staff updated the final ISG so that the section titled "Minimum Scope of Probabilistic Risk Assessment and Alternative Risk Evaluations for a Construction Permit Application" is revised to include the following: *If a PRA is used to support an LWR CP application under 10 CFR Part 50, a reviewer can*

Comment Identifier	Topic	Specific Comment	NRC Staff Response
			refer to table 1 for a summary of the minimum scope of the PRA. However, the scope of the PRA may need to be greater, depending on the intended uses of PRA information for a given application. A reviewer can refer to table 2 for a summary of additional PRA and alternative risk evaluations that can be used to support a CP application. **** *The staff should review the applicant's justification that the scope and level of detail of any PRA or alternative risk evaluation are consistent with the intended uses of the information from those assessments to support the CP application. The staff should review the applicant's plan for assessing any risk contributors not addressed by a PRA or alternative risk evaluation.* This guidance document applies to CP applications. Regulations for CP applications are described in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50. Regulations for early site permits, standard design certifications, combined license, standard design approvals, and manufacturing licenses continue to be described in 10 CFR Part 52, which does not provide for CP applications. Staff guidance is contained in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (SRP).
NRC-2024-0217-DRAFT-0001-4		During the design phase, as aspects impacting risk become known, changes are made to improve risk profiles and overall results. Design improvements may result in changes in actual physical locations of equipment and plant layout. Therefore, discussion of intermediate preliminary results of risk evaluations and dominant risk contributors from some of the hazard models (Fire, Flooding, Seismic, etc.) in the design-phase may not be appropriate for the PSAR as design progresses.	The NRC staff disagrees with this comment. The guidance in the ISG accommodates different levels of design maturity at the time of the CP application. The guidance indicates that the staff should consider whether the risk assessment and severe accident information provided is commensurate with the design maturity, the application for which it is intended, and the role the risk assessment results play in the integrated decision-making process. Throughout the ISG, statements appear to the effect that "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." The discussion of intermediate preliminary results of risk evaluations and dominant risk contributors is covered by the inclusion of the identification of assumptions as described in the ISG. To the extent an applicant seeks NRC consideration of risk information developed during the review, the applicant is free to amend the application to do so, although this may affect the review schedule. The staff consider changes to the PRA made after CP issuance in the review of the operating license

Comment Identifier	Topic	Specific Comment	NRC Staff Response
			(OL) application to the extent the OL application relies on that information. No change was made to this final ISG as a result of this comment.
NRC-2024-0217-DRAFT-0001-5	General	The hazard assessments requirements appear to go beyond traditional requirements. Traditionally, design for [safety-related] SR [structures, systems, and components] SSCs to a design basis hazard level set in accordance with traditional guidance (RG 1.76, RG 1.29, RG 1.59) is acceptable.	The NRC staff disagrees with this comment. The ISG does not provide guidance on design requirements or propose alternate design requirements to traditional design requirements; however, the PRA or alternative risk evaluations should be consistent with the design and assumptions. No change was made to this final ISG as a result of this comment.
NRC-2024-0217-DRAFT-0001-6		Use of PRA should not be required to determine licensing-basis events. The Standard Review Plan chapter 15 Events, combined with the additional events required by regulation, should be sufficient.	The NRC staff agrees with this comment. The use of PRA is not required to determine licensing-basis events. The guidance indicates that the staff should consider whether the risk information is commensurate with the application for which it is intended and the role the PRA results play in the integrated decision-making process. If an applicant elects to use a PRA or alternative risk evaluations for the purpose of determining licensing-basis events, then, under the ISG, the staff would consider whether the discussions and descriptions of the risk information included in the PSAR are commensurate with this use. The NRC staff updated the final ISG so that the section titled “Rationale” is modified to remove the following: *(5) determining licensing basis events* The section titled “Rationale” is modified to add the following: *The PRA and alternative risk evaluations may also be used for other purposes, such as a part of or a basis for the determination of licensing-basis events.*
NRC-2024-0217-DRAFT-0001-7	General	Page 8 states that Capability Category I is acceptable at the Construction Permit (CP) stage, however, it should be clarified that some supporting requirements will be not applicable or not reviewed. This will provide consistency with what is stated on Page 5 and is particularly relevant for Supporting Requirements related to data and human reliability analysis.	The NRC staff agrees with this comment. Consistent with the text on page 5, DC/COL-ISG-028 is 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 is needed for a high-level requirement, as those terms are used in the PRA Standard and RG 1.200. In accordance with the guidance in the ISG, the staff would consider this approach in determining the applicability of supporting requirements in industry standards to a CP application for an LWR. The NRC staff updated the final ISG so that the section titled “Minimum Scope of Probabilistic

Comment Identifier	Topic	Specific Comment	NRC Staff Response
			Risk Assessment and Alternative Risk Evaluations for Construction Permit Application," on page 8, is clarified to read: *The staff should consider whether any particular supporting requirement endorsed in industry standards may not be applicable, or cannot be achieved as written, for the CP stage and should consider the applicant's approach in determining the applicability of supporting requirements.*
NRC-2024-0217-DRAFT-0001-8	General	Page 8 states that the staff encourages the use of PRA for hazards that cannot be screened out at the CP stage. However, there may be hazards for which PRA does not offer additional insights at the CP stage, and this should be accounted for in this guidance. Additionally, per the NRC's PRA Policy Statement, PRA should be used consistent with the state of practice. As the state of the practice for hazards is not sufficient for development of a full PRA model, it is inappropriate for the NRC to encourage use of PRA for these hazards.	The NRC staff agrees in part with this comment. The staff agrees that there may be hazards for which a PRA does not offer additional insights at the CP stage. The staff acknowledges that a draft white paper that was publicly released (ML23326A185) encouraged the maximum use of PRA to assess the risk from hazards that cannot be screened out at the CP stage, the draft ISG issued for public comment (ML24192A277) did not include this statement. Nevertheless, the staff disagrees that encouraging the use of PRA when information is available is inconsistent with the PRA Policy Statement (60 FR 42622). In addition to PRA guidance, the ISG provides guidance on alternative risk evaluations for the CP application. No change was made to this final ISG as a result of this comment.
NRC-2024-0217-DRAFT-0001-9	General	NRC recently provided guidance 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," on the scope of PRA expected at the CP phase for applicants following the Licensing Modernization Project (LMP) methodology. LMP relies on PRA to an extent greater than other licensing application approaches, and therefore, should set an upper limit for NRC expectations of a PRA at the CP phase. o RG 1.253 states, "The CP applicant may disposition certain hazards by crediting [design-basis hazard levels] DBHLs in lieu of explicitly modeling these hazards in the PRA or	The NRC staff agrees in part with this comment. The staff agrees that not all hazards need to be modeled in the PRA at the CP stage. For added clarity, the staff revised the following sections to clarify that plant operating state analysis is not a minimum element: "Uses of Probabilistic Risk Assessment Information in a Construction Permit Application" (page 7); "Plant Operating State Analysis" (page 12); and "Low Power and Shutdown" (page 26). The staff disagrees that it is appropriate to compare the LMP approach with other licensing application approaches. For example, the LMP approach allows for risk metrics other than core damage frequency and large release large release, which are used in other licensing approaches. While there is no regulatory requirement for a PRA under 10 CFR Part 50, if a PRA is used to support a PSAR submitted for a CP application, the PRA nonetheless needs to be adequate to the extent the applicant relies on it. As the guidance in the ISG indicates, the staff should consider whether the risk information is commensurate with the application for which it is intended and the role the PRA results play in the integrated decision-making process. This includes the supporting requirements that apply to the

Comment Identifier	Topic	Specific Comment	NRC Staff Response
		accounting for them through a risk-informed supplementary evaluation." Therefore, it is not appropriate for this ISG to suggest that hazard risk evaluations are required, SR hazard design should be sufficient, in line with the guidance in RG 1.253. o Table A-1 of RG 1.253 does not have the Plant Operating State (POS) element in the "minimal" column. Therefore, it is not appropriate for the ISG to set a "Low-power and shutdown risk evaluation" as a minimum requirement. Traditional analysis of lower modes as discussed in the SRP should be acceptable.	capability and functionality of SSCs credited in the design based even in part on PRA results. The current state of practice is to use an internal events, at-power PRA model as the foundation for representing the response of a facility to perturbations from normal operations. The staff disagrees that the use of LMP would "set an upper limit for NRC expectations of a PRA," supporting an LWR CP application under 10 CFR Part 50. While RG 1.253 establishes that the minimum PRA needed for an LMP-based, non-LWR CP application under 10 CFR Part 50 can be an internal events, at-power reactor PRA logic model, the basis for the staff position in RG 1.253 differs from the basis for the guidance in the ISG because the use of PRA in the LMP methodology is an integral aspect of that approach within its constraints (e.g., frequency-consequence criteria) and is well defined for the entire design life cycle. A reviewer may refer to ISG table 1 for the minimum scope of a PRA to support a CP application. A reviewer may refer to ISG table 2 for additional PRA scope elements and acceptable alternative risk evaluations that may be used to support a CP application, consistent with the intended uses of the related risk information. Tables 1 and 2 are independent of each other. Therefore, if an application provides the minimum scope listed in table 1, that does not necessitate or compel an applicant to include the information identified in table 2 in the application. The ISG tables are consistent with RG 1.253, table A-1, and do not set the plant operating state element as a minimum standard. Further, this ISG provides for a range of alternative risk evaluations. The ISG does not limit the types of alternative risk evaluations that may be used as long as an applicant justifies the applicability of the assessment used. This ISG provides guidance on the type of information to submit if such an alternative evaluation is used. The NRC staff updated the final ISG to add clarifications on the use of tables 1 and 2 to address portions of this comment.
NRC-2024-0217-DRAFT-0002-1		**Section**: Pg. 9, Table 2 **Comment/Basis:** Table 2 lists acceptable alternative risk evaluations for a construction permit application. Alternative risk evaluations for many hazards (e.g., internal flood, fire) are not commonly performed. **Recommendation**: Consider addition of example references	The NRC staff disagrees with this comment. The alternative risk evaluation is one option available for providing the content of risk assessment information in a CP application. The NRC staff is familiar with the use of alternative risk evaluations for many hazards. No change was made to this final ISG as a result of this comment.

Comment Identifier	Topic	Specific Comment	NRC Staff Response
		to Table 2, to guide development of these alternative risk evaluations.	
NRC-2024-0217-DRAFT-0002-2		**Section**: Pg. 14, Passive Safety System Reliability **Comment/Basis**: The passive system reliability discussion includes the following statement: "…the uncertainty in their values, as predicted by a best-estimate thermal hydraulics analysis, can be of comparable magnitude to the predicted values themselves." Is there a basis for this quantitative estimate of uncertainty? In the NuScale probabilistic risk assessment, passive reliability estimates are on the order of E-5; an equivalent amount of uncertainty seems excessive for natural forces. **Recommendation:** Revisit the statement in this section and provide basis for the quantitative estimate of uncertainty.	The NRC staff disagrees with this comment. SRP Section 19.0, page 19.0-21 states, "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] T-H analysis, can be of comparable magnitude to the predicted values themselves." This paragraph provides background on passive safety system reliability and does not state that uncertainty is necessarily of comparable magnitude to the predicted values. The ISG provides guidance for staff to review passive safety system reliability for a CP regardless of the magnitude of uncertainty for the specific application. No change was made to this final ISG as a result of this comment.
NRC-2024-0217-DRAFT-0002-3		**Section**: Pg. 28, Regulatory Treatment of Nonsafety Systems for Designs with Passive Safety Systems **Comment/Basis**: As worded, it is implied that nonsafety-related structures, systems, and components (SSC) subject to the regulatory treatment of non-safety systems (RTNSS) process exist in the design. However, some cases may result in identification of no RTNSS nonsafety-related SSC. **Recommendation:** Clarify that this section applies when nonsafety-related SSC subject to RTNSS are present in the design.	The NRC staff agrees with this comment. The staff agrees that some designs may not include SSCs subject to the RTNSS process. The NRC staff will review the applicant's RTNSS evaluation that leads to the determination that no RTNSS SSCs are identified for an individual design. The NRC staff updated the final ISG to add a third sentence to the introductory paragraph of the section titled "Regulatory Treatment of Nonsafety Systems for Designs with Passive Safety Systems," to state: *RTNSS SSCs may not be identified for a particular design following the RTNSS process.*