U.S. NUCLEAR REGULATORY COMMISSION

STANDARD REVIEW PLAN

19.0 PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION FOR NEW REACTORS

REVIEW RESPONSIBILITIES

Primary - Organization responsible for the review of the applicant's probabilistic risk assessment (PRA)

Organization responsible for the review of severe accident design features

Secondary - Technical organizations identified in the Review Interface section of this SRP may be consulted, as needed

I. AREAS OF REVIEW

This SRP section pertains to the staff review of the design-specific PRA for a design certification (DC) and plant-specific PRA for a combined license (COL) application, respectively. This SRP section also pertains to the staff review of the applicant's deterministic evaluation of design features for the prevention or mitigation of severe accidents.

Subsequent to COL issuance, the staff may review the applicant's PRA (or portions thereof) in the context of licensing actions, following the guidance provided in Regulatory Guides 1.174 and 1.200, SRP sections 19.1 and 19.2 (previously SRP Chapter 19), and associated application-specific regulatory guidance and SRP sections, while continuing to address and maintain the validity of the staff findings associated with the PRA and severe accident related licensing basis.

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The scope of a DC review is limited to the design-specific aspects within the scope of the certification. The design-specific PRA developed during the DC stage may not identify site-specific information (e.g., seismic hazards, switchyard and offsite grid configuration) and may not explicitly model all aspects of the design (e.g., balance of plant). Therefore, the applicant's design-specific PRA may include assumptions regarding site parameters and the interfaces with undeveloped aspects of the design. This is acceptable at the DC stage and results in the identification of PRA-based insights that include design, site, and operational assumptions.

The COL reviews correspond to COL application content guidance provided in Regulatory Guide (RG) 1.206, Section C.I.19 on the PRA and the severe accident evaluation. An outline of the final safety evaluation report (FSER) is presented in Appendix A to this SRP section. For a COL application that references a DC, the staff review of the PRA for the COL should focus on the plant-specific aspects of the PRA and site-specific design features that deviate from the referenced DC and the associated differences in risk results and insights. Similar limitation in the scope of the review applies to severe accident evaluations. This review corresponds to RG 1.206, Section C.III.19.

An applicant's FSAR for both a DC or COL application needs to provide a description of the PRA and its results. This requirement is intended to be a qualitative description of insights and uses, as well as some quantitative PRA results, such that the staff can perform the review, ensure risk insights were factored into the design, and make the evaluation findings described in this SRP section. The complete PRA (e.g., models, analyses, data, and codes) will be available for NRC audit. The staff will document any NRC audits performed in audit reports so that they may be referenced in the staff's safety evaluation report (SER).

There are two aspects of the review. The first aspect is the use of the PRA and severe accident evaluation to identify and assess the balance of preventive and mitigative features, including consideration of operator actions, such that the plant's operation will reflect a reduction in risk compared to existing operating plants.[1] The second aspect is the use and application of the PRA results and insights to support other programs.

1. The applicant's PRA and severe accident evaluation are used as follows:

   A. During the design phase

      i. Identify and address potential design features and plant operational vulnerabilities, where a small number of failures could lead to core damage, containment failure, or large releases (e.g., assumed individual or common-cause failures could drive plant risk to unacceptable levels with respect to the Commission's goals, as presented below),

      ii. Reduce or eliminate the significant risk contributors of existing operating plants that are applicable to the new design by introducing appropriate features and requirements, and

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[1] The reference to existing operating plants applies to light-water reactor (LWR) plant technology contemporary with the issuance of the Commission's Severe Reactor Accident Policy Statement on August 8, 1985.

iii. Select among alternative features, operational strategies, and design options.

B. Identify risk-informed safety insights based on systematic evaluations of the risk associated with the design, construction, and operation of the plant such that the applicant can identify and describe the following:

i. The design's robustness, levels of defense-in-depth, and tolerance of severe accidents initiated by either internal or external events, and

ii. The risk significance of specific human errors associated with the design, and characterize the significant human errors in preparation for better training and more refined procedures.

C. Demonstrate how the risk associated with the design compares against the Commission's goals[2] of less than $1x10^{-4}$/year for core damage frequency and less than $1x10^{-6}$/year for large release frequency.[3] In addition, compare the design against the Commission's approved use of a containment performance goal, which includes (1) a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges and (2) a probabilistic goal that the conditional containment failure probability be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA.

D. Assess the balance of preventive and mitigative features of the design, including consistency with the Commission's guidance in SECY-93-087 and the associated staff requirements memorandum (SRM).

E. Demonstrate whether the plant design, including the impact of site-specific characteristics, represents a reduction in risk compared to existing operating plants.

F. Demonstrate that the design addresses known issues related to the reliability of core and containment heat removal systems at some operating plants (i.e., the additional TMI-related requirements in 10 CFR 50.34(f)).

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2 These are goals and not regulatory requirements, and applicants should not artificially (or intentionally) increase PRA results associated with one metric simply to meet the goal associated with another metric. Rather, the applicant should compare its plant-specific PRA results and insights against these goals and address how its plant features properly balance severe accident prevention and mitigation, consistent with Item D.

3 The Commission staff requirements memorandum (SRM) dated June 26, 1990, in response to SECY-90-016 established the identified goals.

2. The results and insights of the PRA are used to support other programs as follows:

   A. Support the process used to demonstrate whether the regulatory treatment of nonsafety systems (RTNSS) is sufficient and, if appropriate, identify the SSCs included in RTNSS.

   B. Support, as a minimum, regulatory oversight processes (e.g., Mitigating Systems Performance Index, significance determination process) and programs that are associated with plant operations (e.g., technical specifications, reliability assurance, human factors, maintenance rule implementation).

   C. Identify and support the development of specifications and performance objectives for the plant design, construction, inspection, and operation, such as inspections, tests, analyses, and acceptance criteria (ITAAC); RAP; technical specifications; and COL action items and interface requirements.

To ensure that the DC or COL application adequately addresses these aspects, the staff reviews the following:

1. The scope, level of detail, and technical adequacy of the design-specific and plant-specific PRA to ensure that it is appropriate for the DC and COL, respectively, and any identified uses and risk-informed applications.

2. The assumptions in the PRA made during design development/certification, in which a specific site may not have been identified or all aspects of the design (e.g., balance of plant) may not have been fully developed, to confirm that these assumptions are identified in the DC application and either remain valid or are adequately addressed within the COL application.

3. The design features used to prevent or mitigate severe accidents identified within SECY-93-087.

4. As part of the PRA description, how the PRA will be maintained and upgraded to ensure that the PRA, consistent with its identified uses and risk-informed applications, should continue to reasonably reflect the as-designed, as-built, and as-operated plant, including the corrective action and feedback mechanisms involving the periodic evaluation of the PRA on the basis of actual plant-specific equipment, train, and system performance and relevant industry operational experience.

<u>Review Interfaces</u>

All technical branches may use the applicant's PRA results and insights in determining the appropriate scope and depth of their technical review. In addition, the QA staff will use the PRA results and insights in reviewing the applicant's RAP. Therefore, the PRA reviewers should also evaluate the following aspects:

1. Probabilistic and other risk assessment methods used for identifying and prioritizing SSCs based on risk significance

2. The SSCs that are determined by probabilistic and other risk assessment methods to be significant contributors to plant safety

3. The industry operating experience and reliability databases used to identify the significant failure modes

Other technical branches that use the PRA and severe accident evaluation results and insights in their programs, processes, and reviews (e.g., human factors, training, and inspection organizations for addressing significant human actions; emergency preparedness; security; reactor oversight processes; technical specifications; RTNSS; maintenance rule implementation) may need to interface with the PRA staff in evaluating these programs, processes, and reviews.

The technical branches that are responsible for the review of the design of the plant for external events, including external hazards (e.g., seismic, tornado, high wind, and external fire and flood) and in-plant area hazards (e.g., internal fire and flood) may need to support the PRA staff in reviewing these hazards. For example, the technical branch responsible for reviewing the seismic capacity/fragility of SSCs and the site-specific seismic hazard may need to review these aspects of the seismic risk analyses.

The technical branches that perform the systems and T-H analyses may need to interface with the PRA staff reviewers to ensure that the applicant's PRA properly considers and addresses important issues (e.g., failure mechanisms, system interactions, and T-H modeling and uncertainties) and, to the extent possible, to focus their review resources more heavily on areas of higher risk importance.

Some technical branches may need to participate in the review of the applicant's identified risk-informed applications of the PRA (e.g., risk-informed inservice inspection, risk-informed inservice testing, 10 CFR 50.69 implementation, implementation of risk-informed/performance-based fire protection). In addressing these risk-informed programs, the applicant should provide the program description in the appropriate section of its FSAR, with a cross-reference to that program in Chapter 19 of its FSAR. Likewise, the staff should present its review of the program in the appropriate section of the staff evaluation, with a brief description of the application of the PRA and a cross-reference to the subject program evaluation in Chapter 19 of the staff SER.

II. ACCEPTANCE CRITERIA

Requirements

Acceptance criteria are based on meeting the relevant requirements of the following Commission regulations:

For a DC

1. 10 CFR 52.47(8) - The information necessary to demonstrate compliance with any technically relevant portions of the Three Mile Island requirements set forth in 10 CFR 50.34(f), specfically 10 CFR 50.34(f)(1)(i).

2. 10 CFR 52.47(a)(23) - For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, e.g., challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass.

3. 10 CFR 52.47(a)(27) - A description of the design-specific probabilistic risk assessment (PRA) and its results.

For a COL

4. 10 CFR 52.79(a)(17) - Information with respect to compliance with a number of the technically relevant positions of the Three Mile Island requirements in 10 CFR 50.34(f), specfically 10 CFR 50.34(f)(1)(i).

5. 10 CFR 52.79(a)(38) - For light-water reactor designs, a description and analysis of design features for the prevention and mitigation of severe accidents, *e.g.*, challenges to containment integrity caused by core-concrete interaction, steam explosion, high-pressure core melt ejection, hydrogen combustion, and containment bypass.

6. 10 CFR 52.79(a)(46) - A description of the plant-specific PRA and its results.

7. 10 CFR 52.79(c)(1), (d)(1), and (e)(1) - If a COL application references a standard design approval, standard DC, or the use of one or more manufactured nuclear power reactors licensed under Subpart F of 10 CFR Part 52, then, the plant-specific PRA information must use the PRA information for the design approval, design certification, or manufactured reactor, respectively, and must be updated to account for site-specific design information and any design changes or departures.

The PRA staff review should also support (1) the expectation, as stated in 10 CFR 52.47(a)(2) and 10 CFR 52.79(a)(2), that reactors will reflect through their design, construction, and operation an extremely low probability of accidents that could result in the release of significant quantities of radioactive fission products and (2) the objective, as stated in 10 CFR 52.47(a)(4) and 10 CFR 52.79(a)(5), to assess the risk to public health and safety resulting from facility operation and ensure the adequacy of plant SSCs that are provided to prevent accidents and mitigate their consequences.

SRP Acceptance Criteria

Specific SRP acceptance criteria acceptable to meet the relevant requirements of the NRC's regulations identified above are as follows for the review described in this SRP section. The SRP is not a substitute for the NRC's regulations, and compliance with it is not required. However, an applicant is required to identify differences between the design features, analytical techniques, and procedural measures proposed for its facility and the SRP acceptance criteria and evaluate how the proposed alternatives to the SRP acceptance criteria provide acceptable methods of compliance with the NRC regulations.

1. NRC Policy Statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," 50 FR 32138, August 8, 1985.

2. NRC Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants," 51 FR 28044, August 4, 1986.

3. NRC Policy Statement, "Nuclear Power Plant Standardization," 52 FR 34884, September 15, 1987.

4. NRC Policy Statement, "Regulation of Advanced Nuclear Power Plants," 59 FR 35461, July 12, 1994.

5. NRC Policy Statement, "The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," 60 FR 42622, August 16, 1995.

6. SECY-90-016, "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," ADAMS Accession No. ML003707849, January 12, 1990, and the related staff requirements memorandum (SRM), ADAMS Accession No. ML003707885, June 26, 1990.

7. SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," ADAMS Accession No. ML003708021, April 2, 1993, and the related SRM, ADAMS Accession No. ML003708056, July 21, 1993.

8. SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," ADAMS Accession No. ML003708224, June 12, 1996, and the related SRM, ADAMS Accession No. ML003708192, January 15, 1997.

9. SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," ADAMS Accession No. ML003708316, February 18, 1997, and the related SRM, ADAMS Accession No. ML003708232, June 30, 1997.

The first five NRC policy statements provide guidance regarding the appropriate course of action to address severe accidents and the use of PRA. The Commission SRMs relating to SECY-90-016, SECY-93-087, SECY-96-128, and SECY-97-044 provide Commission-approved guidance for implementing features in new designs to prevent severe accidents and to mitigate their effects, should they occur.

For the first aspect of the review, the staff's acceptance criteria consists of a determination that the applicant has adequately demonstrated that the design properly balances preventive and mitigative features and represents a reduction in risk when compared to existing operating plants.

For the second aspect of the review, the staff should ensure that the applicant has used the PRA results and insights, including those from uncertainty analyses, importance analyses, and sensitivity studies, in an integrated fashion to identify and establish specifications and performance objectives (e.g., ITAAC, technical specifications, RAP, RTNSS, and COL action items) for the design, construction, testing, inspection, and operation of the plant. The specific programs establish the staff's acceptance criteria. For example, Section C.I.17.4 of Regulatory Guide 1.206 presents the RAP submittal guidance and SRP Section 17.4 gives the associated staff review guidance, including acceptance criteria.

For designs that have evolved from current plant technology through the incorporation of several features intended to make the plant safer, more available, and easier to operate, the results of the PRA should indicate that the design represents a reduction in risk compared to existing operating plants. The staff review should include a broad (qualitative and quantitative) comparison of risks, by initiating event category, between the proposed design and existing operating plant designs (from which the proposed design evolved) to identify the major design features that contribute to the reduced risk of the proposed design compared to existing plant designs (e.g., passive systems, less reliance on offsite and onsite power for accident mitigation, and divisional separation).

The staff review should also consider the impact of data uncertainties on the risk estimates. The uncertainty analysis should identify major contributors to the uncertainty associated with the estimated risks. In addition, the staff review should address the applicant's risk importance studies that are performed at the system, train, and component level to provide insights about (1) the systems that contribute the most in achieving the low risk level assessed in the PRA, (2) events (e.g., component failures or human errors) that contribute the most to decreases in the built-in plant safety level, and (3) events that contribute the most to the assessed risk. The staff should also review the applicant's sensitivity studies performed to gain insights about the impact of uncertainties (and the potential lack of detailed models) on the estimated risk. The objectives of the sensitivity studies should include (1) determining the sensitivity of the estimated risk to potential biases in numerical values, such as initiating event frequencies, failure probabilities, and equipment unavailabilities, (2) determining the impact of the potential lack of modeling details on the estimated risk, and (3) determining the sensitivity of the estimated risk to previously raised issues (e.g., motor-operated valve reliability).

For designs using passive safety systems and active defense-in-depth systems, the staff should review the sensitivity studies performed to investigate the impact of uncertainties on the PRA results under the assumption of plant operation without credit for the nonsafety-related defense-in-depth systems. These studies provide additional insights about the risk importance of the defense-in-depth systems that are taken into account in selecting nonsafety-related systems for regulatory treatment according to the RTNSS process.

To have confidence that the applicant's PRA and severe accident evaluation results and insights are adequate, the PRA staff must also determine that the scope, level of detail, and technical adequacy of the design-specific and plant-specific PRA are appropriate for the DC and COL, respectively, and any identified uses and risk-informed applications, as follows:

1. The applicant's analyses should be comprehensive in scope, and address all applicable internal and external events and all plant operating modes. Since some aspects of the applicant's approach may involve non-PRA techniques to address specific events (e.g., PRA-based seismic margins), the PRA staff review should ensure that the scope of the applicant's analyses is appropriate for their identified uses and applications, which may involve a scope, level of detail, and/or technical adequacy for the affected areas that is greater than that needed for a COL application.

2. The level of detail of the applicant's PRA should be commensurate with the identified uses and applications of the PRA (e.g., sufficient to gain risk-informed insights and use such insights, in conjunction with assumptions made in the PRA, to identify and support requirements important to the design and plant operation). The PRA should reasonably

reflect the actual plant design, construction, operational practices, and relevant operational experience of the applicant and the industry. The burden is on the applicant to justify that the PRA approach, methods, and data, as well as the requisite level of detail necessary for the NRC staff's review and assessment, are appropriate. Regulatory Guides 1.174 and 1.200 provide additional guidance on the level of detail that should be included in the PRA. If detailed design information (e.g., regarding cable and pipe routing) is not available or if it can be shown that detailed modeling does not provide significant additional information, it is acceptable to make bounding-type assumptions consistent with the guidelines in Regulatory Guide 1.200. However, the risk models should still be able to identify vulnerabilities as well as design and operational requirements such as ITAAC and COL action items. In addition, the bounding assumptions should not mask any risk-significant information about the design and its operation.

3. Consistent with the guidance in Section 2.5 of Regulatory Guide 1.174 regarding QA, the staff expects that the applicant will have subjected its PRA to quality control. The following methods are acceptable to the NRC staff to ensure that the pertinent QA requirements of Appendix B to 10 CFR Part 50 are met and that the PRA is sufficient:

   A. Use of personnel qualified for the analysis

   B. Use of procedures that ensure control of documentation, including revisions, and provide for independent review, verification, or checking of calculations and information used in the analyses

   C. Documentation and maintenance of records, including archival documentation as well as submittal documentation

   D. Use of procedures that ensure that appropriate attention and corrective actions are taken if assumptions, analyses, or information used previously are changed or determined to be in error

   Toward this end, the applicant's PRA submittal should be consistent with prevailing PRA standards, guidance, and good practices as needed to support its uses and applications and as endorsed by the NRC (e.g., Regulatory Guide 1.200 and SRP Section 19.1).[4]

   In addressing the technical adequacy of the PRA, the applicant should include (1) a discussion of prior NRC staff review of the PRA (e.g., during the DC process), findings (i.e., facts and observations) from that review, disposition of those findings, and the relevance of that review to the technical adequacy of the current plant-specific PRA, (2) a discussion of the scope, level of detail, and technical adequacy needed to support

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[4] The applicant's adherence to the recommendations provided in Regulatory Guides 1.174 and 1.200 pertaining to quality and technical adequacy will result in a more efficient and consistent NRC staff review process. Alternatively, the applicant may identify, and justify the acceptability of, alternative measures for addressing PRA quality and technical adequacy, and the staff should specifically review the acceptability of these alternative measures in the context of the specific uses and applications of the PRA.

the specific uses and risk-informed applications, (3) a discussion regarding the method used for determination of technical adequacy for pertinent PRA scope areas for which the NRC has not endorsed PRA standards (i.e., identify the guidance and good practices documents relied upon to determine the technical adequacy of the PRA), (4) a discussion on the use of and criteria for independent peer reviews, and (5) a discussion on the process for dispositioning independent peer review findings and maintaining or upgrading the PRA, as appropriate, to ensure that it reasonably reflects the as-designed, as-built, and as-operated plant, including the corrective action and feedback mechanisms involving the periodic evaluation of the PRA, consistent with its uses and risk-informed applications, on the basis of actual plant-specific equipment, train, and system performance and relevant industry operational experience.

As noted in Element 1.1 of Table A-1 in Appendix A to Regulatory Guide 1.200, special emphasis should be placed on PRA modeling of novel and passive features in the design, as well as addressing issues related to those features, such as digital instrumentation and control, explosive (squib) valves, and the issue of T-H uncertainties.[5]

The staff should confirm that the assumptions made in the applicant's PRA during design development/certification, in which a specific site may not have been identified or all aspects of the design (e.g., balance of plant) may not have been fully developed, are identified in the DC application and either remain valid or are adequately addressed within the COL application.

In addition, a DC and COL applicant may request NRC approval to implement one or more risk-informed applications. The applicant's submission and staff review of these risk-informed applications follow specific regulatory guidance, approved topical reports, and SRP sections. For example, if an applicant requests to implement a risk-informed inservice inspection program concurrent with its COL application, the application should address the guidance provided in Regulatory Guide 1.178, following a specific methodology contained in identified approved topical reports and approved industry code cases, and SRP Section 3.9.8 will guide the staff review of this program. Chapter 19 of the applicant's FSAR should identify this risk-informed application, with a cross-reference to Section 3.9.8 of the applicant's FSAR, which should describe the applicant's risk-informed inservice inspection program.

III. REVIEW PROCEDURES

The reviewer will select material from the procedures described below, as may be appropriate for a particular case.

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[5] The issue of T-H uncertainties arises from the "passive" nature of safety-related systems used for accident mitigation. 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" T-H analysis, can be of comparable magnitude to the predicted values themselves. Therefore, some accident sequences with a frequency high enough to impact results, but which are not predicted to lead to core damage by a best-estimate T-H analysis, may actually lead to core damage when T-H uncertainties are considered in the PRA models.

These review procedures are based on the identified SRP acceptance criteria. For deviations from these acceptance criteria, the staff should review the applicant's evaluation of how the proposed alternatives provide an acceptable method of complying with the relevant NRC requirements identified in Subsection II.

The staff will review the description of the PRA and its results in order to make the evaluation findings described in this SRP section. In addition to a qualitative description, the staff will review some quantitative results (e.g., mean core-damage frequencies, mean large release frequencies, and system importance measures), with the understanding that the complete PRA, including models, analyses, data, and codes, will be available for NRC audit. Although Chapter 19 of the applicant's FSAR should include much of the information needed by the staff to perform the PRA review, some staff audits of the PRA and supporting analyses may be necessary to fully understand, review, and confirm the PRA results, insights, and associated analytical bases. The staff will document any PRA audits performed in audit reports so that they may be referenced in the staff's SER. For instances in which additional information is needed to complete the staff's review of the FSAR, the staff will use the request for additional information (RAI) process.

Through the review of information provided by the applicant and of applicable NRC audit reports, the staff will assess the applicant's PRA and severe accident evaluations. The reviewer's assessment of the PRA, severe accident evaluations, and any commitments lead to the conclusions regarding acceptability, as described in Subsection IV of this SRP section. For additional information, the staff should consider the information provided in Chapter 19 of past SERs for advanced LWRs. The NRC issued the final SER (FSER) for the DC of the advanced boiling-water reactor design as NUREG-1503, the FSER for the DC of the System 80+ reactor design as NUREG-1462, the FSER for the DC of the AP600 design as NUREG-1512, and the FSER for the DC of the AP1000 as NUREG-1793.

The staff reviews the applicant's use of the results and insights of its PRA and severe accident evaluations to establish specifications and performance objectives for plant design, construction, inspection, and operation. The NRC should review this information to ensure that it is able to conclude that the applicant has performed sufficiently complete and scrutable analyses, the results and insights support the application, and the applicant has in place programs and processes that will enable it to maintain an up-to-date PRA for these uses and applications.

As expressed in the review interfaces portion of Subsection I of this SRP section, other technical branches are expected to use the applicant's PRA results and insights to inform their review based on risk significance. This can help define the depth of their technical review, highlight assumptions regarding SSC and operator performance and reliability, ITAAC, interface requirements, plant features, design and operational programs, and other aspects. Working with the licensing project organization, the PRA staff will provide the applicant's PRA results and insights, specifically the relative risk ranking of SSCs and operator actions as well as identifying assumptions and limitations in the analyses that may impact these results and insights to the other technical branches. As the staff reviews the applicant's PRA information, if the applicant's PRA results and insights change significantly, the staff through the licensing project organization should convey these changes to the other technical branches to ensure that the scope and depth of their reviews are properly focused and at the appropriate level of detail.

1. Specific Review Guidance

The staff should ensure that the applicant's design-specific/plant-specific PRA and severe accident evaluations address the aspects identified in Subsection I of this SRP section and address the related requirements in 10 CFR Part 52, as well as the related Commission policies and positions identified in Subsection II.

The staff reviews the process used by the applicant to develop its design-specific/plant-specific PRA, including supporting or associated analyses (e.g., T-H analyses, human reliability analyses). The staff also reviews the applicant's process for maintaining and upgrading the PRA, as necessary, to ensure that (1) it reasonably reflects the plant design, operation, and experience and (2) its scope, level of detail, and technical adequacy are appropriate for its uses and risk-informed applications.

The staff should review the applicant's uses and risk-informed applications (or proposed uses and applications)[6] of the PRA for each phase (i.e., design, licensing, construction, operations). Specifically, the PRA staff should interface with the primary review branch in the evaluation of licensee programs that use the PRA results and insights (e.g., human factors program, severe accident management program, maintenance rule implementation, interface with the reactor oversight process, RAP). Likewise, the PRA staff should interface with the primary review branch in the evaluation of specific risk-informed applications (e.g., 10 CFR 50.69 implementation, risk-informed performance-based fire protection implementation, risk-informed inservice inspection, risk-informed inservice testing). When the applicant enters a new phase, the staff review may include evaluating any changes to the uses and applications (or proposed uses and applications of subsequent phases).

The PRA staff should review the applicant's PRA and its associated results and insights, addressing the full scope of operations (full power, low power, shutdown), as discussed below.

A. Internal Events at Full Power

For internal events occurring under full-power operating conditions, this review should address the PRA results and insights of the plant-specific PRA. This review should ensure that the internal events are properly evaluated, including

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[6] If the applicant is merely identifying expected uses and applications of the PRA in subsequent phases, the staff will typically not perform detailed reviews of this information, but rather will use this information in planning and preparing for these reviews in the subsequent phases. However, if the applicant requests, as part of its application, review and approval of a risk-informed application that will be implemented during a later phase, the staff will perform the necessary review. For example, at the COL phase, an applicant may propose to implement 10 CFR 50.69 during the construction phase and need to have that risk-informed application approved before that time.

evaluating the appropriateness of internal events that are screened out or incorporated into other evaluations (e.g., grouped events). For the internal events evaluated, the staff should review the PRA to ensure that it appropriately identifies and describes the following aspects:

i. Significant Core Damage Sequences and their numerical values

- Significant[7] core damage sequences
- Significant internal initiating events
- Significant functions, SSCs, and operator actions (typically determined by importance measures such as risk achievement worth and Fussell-Vesely importance measures)
- PRA assumptions[8] and PRA-based insights[9]
- Results and insights from importance, sensitivity, and uncertainty analyses

ii. Significant Large Release Sequences and their numerical values

- Significant large release sequences
- Significant internal initiating events
- Significant functions, SSCs, and operator actions
- Containment performance
- PRA assumptions and PRA-based insights
- Results and insights from importance, sensitivity, and uncertainty analyses

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7 In the context of the PRA results and insights, the term “significant” is intended to be consistent with its usage in the American Society of Mechanical Engineers (ASME) PRA Standard, ASME RA-Sb-2005 Addenda to ASME RA-S-2002.

8 In the context of the PRA, the phrase “assumption” is intended to be consistent with its usage in RG 1.200.

9 “PRA-based insights” are those insights identified during the DC process that ensure that assumptions made in the PRA will remain valid in the as-to-be-built, as-to-be-operated plant and include assumptions regarding SSC and operator performance and reliability, ITAAC, interface requirements, plant features, design and operational programs, and others. The usage of this phrase is intended to be consistent with its use in referring to the information provided in Table 19.59-29 in the AP600 design control document.

B. External Events at Full Power

For external events occurring under full-power operating conditions, this review should address the PRA results and insights of the plant-specific PRA. This review should ensure that the external events are properly evaluated, including evaluating the appropriateness of external events that are screened out, bounded, or incorporated into other evaluations (e.g., grouped events). For each external event evaluated, the staff should review the PRA to ensure that it appropriately identifies and describes the following aspects:

i. Significant Core Damage Sequences and their numerical values

- Significant core damage sequences
- Significant external initiating events
- Significant functions, SSCs, and operator actions
- PRA assumptions and PRA-based insights
- Results and insights from importance, sensitivity, and uncertainty analyses

ii. Significant Large Release Sequences and their numerical values

- Significant large release sequences
- Significant external initiating events
- Significant functions, SSCs, and operator actions
- Containment performance
- PRA assumptions and PRA-based insights
- Results and insights from importance, sensitivity, and uncertainty analyses

C. Events During Other Operating Modes

For events occurring under conditions other than full power (e.g., low power and shutdown), this review should address the PRA results and insights of the plant-specific PRA. This review should ensure that the modes are properly evaluated, including evaluating the appropriateness of modes that are screened out, bounded, or incorporated into other modes (e.g., grouped modes). For each mode (other than full power) evaluated, the staff should review the PRA to ensure that it appropriately identifies and describes the following aspects:

i. Significant Core Damage Sequences and their numerical values

- Significant core damage sequences
- Significant initiating events
- Significant functions, SSCs, and operator actions
- PRA assumptions and PRA-based insights
- Results and insights from importance, sensitivity, and uncertainty analyses

ii. Significant Large Release Sequences and their numerical values

- Significant large release sequences
- Significant initiating events
- Significant functions, SSCs, and operator actions
- Containment performance
- PRA assumptions and PRA-based insights
- Results and insights from importance, sensitivity, and uncertainty analyses

The PRA staff should also review the overall/integrated results and insights from the design-specific/plant-specific PRA. In particular, the staff should review the applicant's identification of significant plant features (including nonsafety-related systems) and operator actions that are important to reduce risk and confirm that the plant will meet the expectation stated in 10 CFR 52.79(a)(2) 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." In addition, for a COL referencing a DC, the staff should compare the COL applicant's PRA-based insights table with the table from the DC to identify potential changes that could impact the PRA results and insights and to verify that the applicant has properly dispositioned each insight.

SRP Section 19.1, which is associated with Regulatory Guide 1.200, provides additional information regarding the review of the technical adequacy of a design-specific/plant-specific PRA. SRP Section 19.1 specifically provides review guidance on determining PRA technical adequacy, and Regulatory Guide 1.200 presents an acceptable approach for determining that the quality of the applicant's PRA is sufficient to provide confidence in the results. In particular, Section C.1 of Regulatory Guide 1.200 describes the functional requirements of a technically acceptable PRA, which may be useful to the staff in determining the technical acceptability of the applicant's design-specific/plant-specific PRA.

In addition, although it relates to the staff review of license applications requesting risk-informed changes to an applicant's licensing bases, SRP Section 19.2 provides specific review guidance in Subsection IV and Appendix A for specific aspects of a PRA (e.g., scope, initiating event analysis, accident sequence analysis) that would be useful to the staff in performing the reviews of new reactor designs for DC and licensing. In particular, Section A.11 of SRP Section 19.2 lists documents that the staff could use as reference or background material during the review process. The bibliography lists documents by category, covering desirable PRA attributes, review of the PRA, uncertainty and sensitivity analyses, and use of the PRA in risk ranking. It also provides a bibliography for aspects of PRA modeling (e.g., initiating events, common-cause failure modeling, human performance modeling).

2. Severe Accident Evaluations

The staff should review the applicant's description and analysis of the design features to prevent and mitigate severe accidents, in accordance with the requirements in 10 CFR 52.47(23) or 10 CFR 52.79(a)(38), for a DC or a COL application, respectively. This review should specifically address the issues identified in SECY-90-016 and SECY-93-087, which the Commission approved in related SRMs dated June 26, 1990, and July 21, 1993, respectively, for prevention (e.g., anticipated transients without scram, mid-loop operation, station blackout, fire protection, and intersystem loss-of-coolant accident) and mitigation (e.g., hydrogen generation and control, core debris coolability, high-pressure core melt ejection, containment performance, dedicated containment vent penetration, equipment survivability).

In addition, the review should address the information provided by the applicant to satisfy the requirements of 10 CFR 52.47(8) or 10 CFR 52.79(a)(17), for a DC or a COL application, respectively. In particular, both regulations invoke 10 CFR 50.34(f)(1)(i) to specify that a design-specific or plant-specific PRA should be performed to seek improvements in core heat removal system reliability and containment heat removal system reliability that are significant and practical and do not excessively impact the plant.

3. PRA-Related ITAAC, COL Action Items, and Other Commitments

The staff should review the PRA-related ITAAC, COL action items, and other commitments, including any actions identified or proposed to address them. The applicant's PRA-based insights table should also identify these items. The staff should note any item that cannot be resolved until after the COL application phase and review the commitments and schedule for resolution of the given items.

IV. EVALUATION FINDINGS

The reviewer verifies that the applicant has provided sufficient information and that the review and calculations (if applicable) support conclusions of the following type to be included in the staff's safety evaluation report. The reviewer also states the bases for those conclusions.

Following the format and content guidance provided in Appendix A to this SRP section will enhance the efficiency and consistency of the staff review documentation. Deviations from the format may occur for review areas that are not relevant for the specific application and are acceptable if the deviation supports better clarity in understanding the staff evaluation.

The staff should provide a summary description of the applicant's design-specific/plant-specific PRA and severe accident evaluations. It should also identify the PRA and severe accident evaluation information that the applicant docketed and the information reviewed by the NRC during audits. The results of the staff review, including staff audits, should reflect a consistent and scrutable evaluation of the applicant's PRA and severe accident evaluations. To reach a finding of acceptability, reviewers will generally need to find the following:

1. The PRA and severe accident evaluation aspects identified in Subsection I of this SRP section are adequately addressed.

2. The PRA reasonably reflects the as-designed, as-built, and as-operated plant, and the PRA maintenance program will ensure that the PRA will continue to reflect the as-designed, as-built, and as-operated plant, consistent with its identified uses and applications.

3. The PRA is of the appropriate scope, level of detail, and technical adequacy for its identified uses and applications.

The staff should support findings of acceptability with logical bases built from an evaluation of the considerations given in Subsection III of this SRP section. Reviewers should verify that the applicant provided sufficient information to complete the review in accordance with this SRP section and therefore that the review is sufficiently complete to support its general findings as identified above, which should be included in the staff's SER.

SRP Sections 19.1 and 19.2 provide additional information regarding the review and evaluation findings for specific aspects of a design-specific/plant-specific PRA. Specifically, SRP Section 19.1 provides review guidance on determining PRA technical adequacy. Although it relates to the staff review of license applications requesting risk-informed changes to the licensing bases, SRP Section 19.2 provides specific review guidance in Subsection IV and Appendix A for specific aspects of a PRA (e.g., scope, initiating event analysis, accident sequence analysis) that would be helpful to the staff in performing the reviews of new reactor designs for DC and licensing.

For DC and COL reviews, the findings will also summarize (to the extent that the review is not discussed in other SER sections) the staff's evaluation of the ITAAC, including design acceptance criteria, as applicable, and interface requirements and combined license action items relevant to this SRP section.

V. IMPLEMENTATION

The staff will use this SRP section in performing safety evaluations of DC applications and license applications submitted by applicants pursuant to 10 CFR Part 52. Except when the applicant proposes an acceptable alternative method for complying with specified portions of the Commission's regulations, the staff will use the method described herein to evaluate conformance with Commission regulations.

The provisions of this SRP section apply to reviews of applications submitted six months or more after the date of issuance of this SRP section, unless superseded by a later revision.

VI. REFERENCES

1. Regulatory Guide 1.174, Revision 1, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." July 2002.

2. Regulatory Guide 1.200, Revision 1, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities." January 2007.

3. Regulatory Guide 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)." June 2007.

4. 10 CFR Part 52, "Early Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

5. U.S. Nuclear Regulatory Commission, Washington D.C. "Severe Reactor Accidents Regarding Future Designs and Existing Plants." *Federal Register*: Volume 50, No. 153. pp 32138-32150. August 8, 1985.

6. U.S. Nuclear Regulatory Commission, Washington D.C. "Safety Goals for the Operations of Nuclear Power Plants Policy Statement." *Federal Register*: Volume 51, No. 149. pp 28044-28049. August 4, 1986.

7. U.S. Nuclear Regulatory Commission, Washington D.C. "Nuclear Power Plant Standardization." *Federal Register*: Volume 52, No. 178. pp 34884-34886. September 15, 1987.

8. U.S. Nuclear Regulatory Commission, Washington D.C. "Regulation of Advanced Nuclear Power Plants: Statement of Policy." *Federal Register:* Volume 59, No. 132. pp 35461-35462. July 12, 1994.

9. U.S. Nuclear Regulatory Commission, Washington D.C. "Use of Nuclear Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement." *Federal Register*: Volume 60, No. 158. pp 42622-42629. August 16, 1995.

10. U.S. Nuclear Regulatory Commission, Washington, D.C. SECY-90-016. "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements." January 12, 1990.

11. U.S. Nuclear Regulatory Commission, Washington, D.C. Staff Requirements Memorandum on SECY-90-016. "Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements." June 26, 1990.

12. U.S. Nuclear Regulatory Commission, Washington, D.C. SECY-93-087. "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," NRC: Washington, D.C. April 2, 1993.

13. U.S. Nuclear Regulatory Commission, Washington, D.C. Staff Requirements Memorandum on SECY-93-087. "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs," NRC: Washington, D.C. July 21, 1993.

14. U.S. Nuclear Regulatory Commission, Washington, D.C. SECY-96-128. "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design." June 12, 1996.

15. U.S. Nuclear Regulatory Commission, Washington, D.C. Staff Requirements Memorandum on SECY-96-128. "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design." January 15, 1997.

16. U.S. Nuclear Regulatory Commission, Washington, D.C. SECY-97-044. "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design." February 18,1997.

17. U.S. Nuclear Regulatory Commission, Washington, D.C. Staff Requirements Memorandum on SECY-97-044. "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design." June 30, 1997.

APPENDIX A

STAFF SAFETY EVALUATION REPORT INPUT FOR PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION

FORMAT AND CONTENT

[This Appendix to SRP Section 19 provides a typical format and content for the staff safety evaluation report. This format and content guidance will enhance the efficiency and consistency of staff review documentation. Deviations from the format may occur for review areas that are not relevant for the specific application and are acceptable if the deviation supports better clarity in understanding the staff evaluation.]

19 PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION

Background
[describe pertinent 10 CFR Part 52 requirements, regulatory guidance, policy statements, SECYs/staff requirements memoranda (SRMs)]

19.1 Probabilistic Risk Assessment

Executive Summary
[at a high level, summarize the NRC review, uses of the probabilistic risk assessment (PRA) in the design process, other uses and risk-informed applications of the PRA, PRA quality, special preventive and mitigative design features, major safety insights from the internal events PRA for operation at power, major safety insights from the external events PRA for operation at power, major safety insights from the PRA for low-power and shutdown operations, and conclusions and findings]

19.1.1 Introduction

19.1.1.1 NRC Review
[present the review scope as identified in Subsection I (Areas of Review) of this SRP section]

19.1.1.2 Uses and Applications of the PRA
[describe the applicant's identified uses and risk-informed applications of the PRA, including discussions, as appropriate, on topics such as the reliability assurance program, regulatory treatment of nonsafety-related systems, Mitigating Systems Performance Index, significance determination process, technical specifications, human factors, maintenance rule implementation, security, emergency planning, risk-informed inservice inspection, risk-informed inservice testing, risk-informed performance-based fire protection implementation, and 10 CFR 50.69 implementation]

19.1.2 Quality of PRA
[discuss PRA quality in the context of the uses and risk-informed applications identified in Section 19.1.1.2]

19.1.2.1 PRA Scope

19.1.2.2 PRA Level of Detail

19.1.2.3 PRA Technical Adequacy

19.1.2.4 PRA Maintenance and Upgrade

19.1.3 Special Design/Operational Features

[address the design and operational features intended to improve plant safety, thus reducing risk when compared to currently operating nuclear power plants; address system interdependencies (e.g., provide system dependency matrix)].

19.1.3.1 Design/Operational Features for Preventing Core Damage
[describe the key preventive features that are intended to minimize initiation of plant transients, arrest the progression of plant transients once they start, and prevent severe accidents (core damage)]

19.1.3.2 Design/Operational Features for Mitigating the Consequences of Core Damage and Preventing Releases from Containment
[describe the key mitigative features that are intended to arrest progression of the core damage event and maintain the integrity of the reactor vessel and containment pressure boundary]

19.1.3.3 Design/Operational Features for Mitigating the Consequences of Releases from Containment
[describe the mitigating features that are intended to terminate releases from containment and minimize offsite doses/consequences]

19.1.3.4 Uses of the PRA in the Design Process
[explicitly address the uses of the PRA in the design process by the applicant (1) introducing features and requirements to reduce or eliminate the known weaknesses/vulnerabilities in current reactor designs, (2) indicating the effect of new design features and operational strategies on plant risk, and (3) identifying and using the PRA-based insights and assumptions to develop design requirements]

19.1.4 Safety Insights from the Internal Events PRA for Operations at Power

[describe insights, including significant initiating events and accident sequences for core damage, releases from containment, and offsite consequences; core damage frequency, large release frequency, conditional containment failure probability, and offsite consequences; identification of areas in which features are the most effective in reducing risk (as compared to current operating plants); significant functions and contributors (e.g., hardware failures; structure, system, and component (SSC) unavailabilities; and human errors) to risk (as defined by core damage, containment releases, and offsite consequences); significant functions and contributors to maintaining plant safety to ensure that risk does not increase unacceptably; significant contributors to the uncertainties associated with the risk estimates; sensitivity of the estimated risk caused by potential biases in numerical values, assumptions, lack of modeling details, and previously raised safety issues; and accident classes that contribute to containment failure]

19.1.4.1 Results and Insights from the Level 1 Internal Events PRA
[include core damage frequency]

19.1.4.1.1 Level 1 PRA Methodology

19.1.4.1.2 Significant Accident Sequences Leading to Core Damage

19.1.4.1.3 Leading Contributors to Core Damage from the Level 1 Internal Events PRA

19.1.4.1.4 Risk-Significant Equipment/Functions/Design Features, Phenomena/Challenges, and Human Actions

19.1.4.1.5 Insights from the Uncertainty, Importance, and Sensitivity Analyses
[include sensitivity to potential biases in numerical values (assumed reliabilities/availabilities), potential lack of modeling details, and previously raised issues]

19.1.4.2 Results and Insights from the Level 2 (Containment Analysis) Internal Events
[include frequency of containment failure (including containment bypass, early containment failure, intermediate containment failure, late containment failure, and containment isolation failure) and conditional containment failure probability]

19.1.4.2.1 Level 2 PRA Methodology
[discuss interface with core damage evaluation (Level 1 PRA), severe accident physical processes/phenomena and modeling, success criteria, accident classes/release categories, and containment ultimate pressure capacity]

19.1.4.2.2 Significant Accident Sequences and Accident Classes Contributing to Containment Failure

19.1.4.2.3 Leading Contributors to Containment Failure from the Level 2 Internal Events PRA
[include separate discussions, as applicable, on containment bypass, early containment failure, intermediate containment failure, late containment failure, and containment isolation failure]

19.1.4.2.4 Risk-Significant Equipment/Functions/Design Features, Phenomena/Challenges, and Human Actions

19.1.4.2.5 Insights from the Uncertainty, Importance, and Sensitivity Analyses

19.1.4.3 Results and Insights from the Level 3 (Offsite Consequences) Internal Events PRA
[if the applicant elected to perform a Level 3 PRA (e.g., to support the evaluation of potential design improvements under 10 CFR 50.34(f)), then include an evaluation of the Level 3 PRA as indicated in Sections 19.1.4.3.1 through 19.1.4.3.5 below; otherwise, Section 19.1.4.3 (including all subsections) may be omitted]

19.1.4.3.1 Level 3 PRA Methodology
[discuss interface with containment analyses (Level 2 PRA), fission product source terms, and dose consequence modeling, including evacuation considerations]

19.1.4.3.2 Significant Accident Sequences and Accident Classes/Release Categories Contributing to Offsite Consequences

19.1.4.3.3 Leading Contributors to Risk from the Level 3 Internal Events PRA

19.1.4.3.4 Risk-Significant Equipment/Functions/Design Features, Phenomena/Challenges, and Human Actions

19.1.4.3.5 Insights from the Uncertainty, Importance, and Sensitivity Analyses

19.1.5 Safety Insights from the External Events PRA for Operations at Power
[describe insights similar to those provided in Section 19.1.4]

19.1.5.1 Results and Insights from the Seismic Risk Evaluation

19.1.5.1.1 Methodology and Approach
[address the applicant's seismic analysis methodology and approach, including any screening and bounding analyses, site seismic hazards analysis, SSC fragility analysis, and accident sequence and system modeling]

19.1.5.1.2 Significant Accident Sequences and Leading Contributors
[address core damage, containment failure, and offsite consequences (if the applicant has elected to perform a Level 3 PRA; see Section 19.1.4.3)]

19.1.5.1.3 Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions

19.1.5.1.4 Insights from the Uncertainty, Importance, and Sensitivity Analyses

19.1.5.2 Results and Insights from the Internal Fires PRA

19.1.5.2.1 Methodology and Approach
[address the applicant's fire analysis methodology and approach, including any screening analyses, fire initiation, fire damage modeling, and plant response analysis and modeling]

19.1.5.2.2 Significant Accident Sequences and Leading Contributors
[address core damage, containment failure, and offsite consequences (if the applicant has elected to perform a Level 3 PRA; see Section 19.1.4.3)]

19.1.5.2.3 Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions

19.1.5.2.4 Insights from the Uncertainty, Importance, and Sensitivity Analyses

19.1.5.3 Results and Insights from Internal Flooding PRA

19.1.5.3.1 Methodology and Approach
[address the applicant's internal flooding analysis methodology and approach, including any screening analyses, flooding sources identification, and flooding evaluation and modeling]

19.1.5.3.2 Significant Accident Sequences and Leading Contributors
[address core damage, containment failure, and offsite consequences (if the applicant has elected to perform a Level 3 PRA; see Section 19.1.4.3)]

19.1.5.3.3 Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions

19.1.5.3.4 Insights from the Uncertainty, Importance, and Sensitivity Analyses

19.1.5.4 <u>Results and Insights from Tornado PRA</u>

19.1.5.4.1 Methodology and Approach
[address the applicant's tornado analysis methodology and approach, including any screening analyses, tornado frequency and magnitude determination, and tornado evaluation and modeling]

19.1.5.4.2 Significant Accident Sequences and Leading Contributors
[address core damage, containment failure, and offsite consequences (if the applicant has elected to perform a Level 3 PRA; see Section 19.1.4.3)]

19.1.5.4.3 Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions

19.1.5.4.4 Insights from the Uncertainty, Importance, and Sensitivity Analyses

19.1.5.*N*[10] <u>Results and Insights from [*Other External Event*] PRA</u>

19.1.5.*N*.1 Methodology and Approach
[address the applicant's [*other external event*] analysis methodology and approach, including any screening analyses, site occurrence frequency and magnitude, and event evaluation and modeling]

19.1.5.*N*.2 Significant Accident Sequences and Leading Contributors
[address core damage, containment failure, and offsite consequences (if the applicant has elected to perform a Level 3 PRA; see Section 19.1.4.3)]

19.1.5.*N*.3 Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions

19.1.5.*N*.4 Insights from the Uncertainty, Importance, and Sensitivity Analyses

19.1.6 Safety Insights from the PRA for Other Modes of Operation
[describe insights for low-power, shutdown, and other than full-power modes of operation, at a level of detail similar to that provided in Section 19.1.4]

19.1.6.1 <u>Results and Insights from Internal Events Low-Power and Shutdown Operations PRA</u>

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[10] *N* is used in this format and content outline as a designator that indicates additional information and associated additional subsections may need to be included to address additional topics (e.g., other external events such as external flooding, tsunamis) as appropriate for the specific plant design and site.

19.1.6.1.1 Significant Accident Sequences and Leading Contributors
[address core damage, containment failure, and offsite consequences]

19.1.6.1.2 Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions
[discuss aspects such as technical specifications, outage planning and control, operator training, and emergency response guidance]

19.1.6.1.3 Insights from the Uncertainty, Importance, and Sensitivity Analyses

19.1.6.2 <u>Results and Insights from External Events Low-Power and Shutdown Operations PRA</u>

19.1.6.2.1 Significant Accident Sequences and Leading Contributors
[address core damage, containment failure, and offsite consequences]

19.1.6.2.2 Risk-Significant Functions/Features, Phenomena/Challenges, and Human Actions

19.1.6.2.3 Insights from the Uncertainty, Importance, and Sensitivity Analyses

19.1.7 PRA-Related Input to Other Programs and Processes
[describe the specific PRA-related inputs provided to the programs identified in Section 19.1.1.2 and provide a cross-reference to the specific sections that describe and evaluate each of these programs]

19.1.7.1 PRA Input to Design Programs and Processes
[discuss those PRA-based insights identified during design development/certification that ensure that assumptions made in the PRA will remain valid in the as-to-be-built, as-to-be-operated plant and include assumptions regarding SSC and operator performance and reliability; inspection, test, analysis, and acceptance criteria; interface requirements; combined license (COL) action Items; plant features, design and operational programs, and other factors]

19.1.7.2 PRA Input to the Maintenance Rule Implementation

19.1.7.3 PRA Input to the Reactor Oversight Process

19.1.7.4 PRA Input to the Reliability Assurance Program

19.1.7.5 PRA Input to the Regulatory Treatment of Nonsafety-Related Systems Program

19.1.7.*N* PRA Input to [*Other Program or Process*]

19.1.8 Conclusions and Findings

19.2 Severe Accident Evaluation

19.2.1 Introduction
[describe the applicant's approach to resolving severe accident issues for the design and the staff determination whether the applicant is consistent with the guidance in SECY-93-087, SECY-96-128, SECY-97-044, and others and corresponding SRMs]

19.2.2 Severe Accident Prevention

19.2.2.1 Severe Accident Prevention Features

19.2.2.1.1 Anticipated Transients Without Scram

19.2.2.1.2 Mid-Loop Operations

19.2.2.1.3 Station Blackout

19.2.2.1.4 Fire Protection

19.2.2.1.5 Intersystem Loss of Coolant Accident

19.2.2.1.*N* [*Other*] Severe Accident Preventive Features

19.2.3 Severe Accident Mitigation

19.2.3.1 Overview of the Containment Design

19.2.3.2 Severe Accident Progression

19.2.3.2.1 In-Vessel Melt Progression

19.2.3.2.2 Ex-Vessel Melt Progression

19.2.3.3 Severe Accident Mitigation Features

19.2.3.3.1 External Reactor Vessel Cooling

19.2.3.3.2 Hydrogen Generation and Control

19.2.3.3.3 Core Debris Coolability

19.2.3.3.4 High-Pressure Melt Ejection

19.2.3.3.5 Fuel-Coolant Interaction

19.2.3.3.5.1 In-Vessel Steam Explosion

19.2.3.3.5.2 Ex-Vessel Steam Explosion

19.2.3.3.6 Containment Bypass

19.2.3.3.6.1 Steam Generator Tube Rupture

19.2.3.3.6.2 Intersystem Loss of Coolant Accident

19.2.3.3.7 Equipment Survivability

19.2.3.3.7.1 Equipment and Instrumentation Necessary to Survive

19.2.3.3.7.2 Severe Accident Environmental Conditions

19.2.3.3.7.3 Basis for Acceptability

19.2.3.3.*N* [*Other*] Severe Accident Mitigation Features

19.2.4 Containment Performance Capability
[address the containment performance goals identified in SECY-93-087 and SECY-90-016, as approved by the associated SRMs—specifically, a deterministic goal that containment integrity be maintained for approximately 24 hours following the onset of core damage for the more likely severe accident challenges, and a probabilistic goal that the conditional containment failure probability be less than approximately 0.1 for the composite of all core damage sequences assessed in the PRA]

19.2.5 Accident Management
[describe those actions taken during the course of an accident by the plant operating and technical staff to (1) prevent core damage, (2) terminate the progress of core damage if it begins and retain the core within the reactor vessel, (3) maintain containment integrity as long as possible, and (4) minimize offsite releases]

19.2.6 Consideration of Potential Design Improvements Under 10 CFR 50.34(f)
[describe the requirement of 10 CFR 50.34(f)(1)(I) that an applicant must “perform a plant/site-specific PRA the aim of which is to seek such improvements in the reliability of core and containment heat removal systems as are significant and practical and do not impact excessively on the plant”]

19.2.6.1 Introduction

19.2.6.2 Estimate of Risk for Design

19.2.6.3 Identification of Potential Design Improvements

19.2.6.4 Risk Reduction Potential of Design Improvements

19.2.6.5 Cost Impacts of Candidate Design Improvements

19.2.6.6 Cost-Benefit Comparison

19.2.6.7 Conclusions

19.3 Open, Confirmatory, and COL Action Items Identified as Unresolved

19.3.1 Resolution of Open Items

19.3.2 Resolution of Confirmatory Items

19.3.3 Resolution of COL Action Items

SRP Section 19.0
Description of Changes

This is the initial issuance of this SRP section.