C.I.19 Probabilistic Risk Assessment and Severe Accident Evaluation

Chapter 19 of the final safety analysis report (FSAR) should provide an adequate level of documentation to enable the NRC staff to determine the acceptability of the risks to public health and safety associated with operation of a proposed new plant. The acceptability of the risks to public health and safety is determined from the interpretation of the results and insights of the applicant's (1) plant-specific probabilistic risk assessment[1] (PRA) and (2) severe accident evaluations.

C.I.19.1 Regulatory Basis

The Commission originally issued 10 CFR Part 52 on April 18, 1989. This rule provided for issuing early site permits (ESPs), standard design certifications, and combined licenses (COLs) with conditions for nuclear power reactors. It stated the review procedures and licensing requirements for applications for these new licenses and certifications and was intended to achieve the early resolution of licensing issues, as well as to enhance the safety and reliability of nuclear power plants. With regard to severe accidents, 10 CFR Part 52 codified some parts of the guidance in the Severe Accident Policy Statement and Standardization Policy Statement.

In 2007, the NRC published a revision to 10 CFR Part 52 and 10 CFR Part 50. The revision to 10 CFR Part 52 included the requirement for a COL applicant to conduct a plant-specific PRA, and to provide a description of the plant-specific PRA and its results within its FSAR. The revision to 10 CFR Part 50 included the requirement for the COL holder to maintain and upgrade the PRA periodically throughout the life of plant.

The NRC has previously issued guidance for addressing PRA and severe accidents in new plant licensing, including the following:

- NRC Policy Statement, "Severe Reactor Accidents Regarding Future Designs and Existing Plants," 50 FR 32138, August 8, 1985.
- NRC Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants," 51 FR 28044, August 4, 1986.
- NRC Policy Statement, "Nuclear Power Plant Standardization," 52 FR 34844, September 15, 1987.
- NRC Policy Statment, "Regulation of Advanced Nuclear Power Plants, 59 FR 35461, July 12, 1994.
- NRC Policy Statement, "The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities," 60 FR 42622, August 16, 1995.

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[1] References in this guide to the plant-specific probabilistic risk assessment (PRA) includes both PRA techniques and alternative approaches for addressing contributors to risk, per the Commission direction provided in the staff requirements memorandum (SRM), dated July 21, 1993, for SECY-93-087.

- 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.

- 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.

- 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 .

- 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 four documents provide guidance regarding the appropriate course of action to address severe accidents and the use of PRA. The 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.

C.I.19.2 Uses of PRA and Severe Accident Evaluations

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[2] that are applicable to the new design by introducing appropriate features and requirements, and

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:

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[2] 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.

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, including a characterization of the significant human errors that may be used as an input to operator training programs and procedure refinement.

C. Demonstrate how the risk associated with the design compares against the Commission's goals 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)).

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., the Mitigating Systems Performance Index (MSPI) and the significance determination process (SDP), and programs that are associated with plant operations, e.g., technical specifications, reliability assurance, human factors, and Maintenance Rule (10 CFR 50.65) implementation.

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[3] 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.

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); the reliability assurance program (RAP); technical specifications; and COL action items and interface requirements.

These uses of the PRA and severe accident evaluation and the uses of the PRA results and insights are drawn from 10 CFR Part 52, the Commission's Severe Reactor Accident Policy Statement regarding future designs and existing plants, the Commission's Safety Goals Policy Statement, the Commission-approved positions concerning severe accidents contained in SECY-93-087, and NRC interest in the use of PRA to help improve future reactor designs.

All uses of the PRA and severe accident evaluation should reflect the potential limitations of the PRA, as indicated by the results of sensitivity and uncertainty analyses.

C.I.19.3 Scope

The applicant's PRA should be a Level 1 and Level 2 PRA that includes internal and external events and addresses all plant operating modes. The scope should be sufficient to enable the applicant to utilize it as discussed in Section C.I.19.2. The scope of the PRA may need to be expanded if it supports other risk-informed applications.[4]

C.I.19.4 Level of Detail

The level of detail of the applicant's PRA should be commensurate with the uses and applications discussed in Section C.I.19.2 (i.e., 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 realistically reflect the actual plant design, planned construction, anticipated operational practices, and relevant operational experience of the applicant and the industry.

The applicant should 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 for the COL application. Additional guidance on the level of detail that should be included in the PRA is provided in Regulatory Guide 1.200, "An Approach For Determining The Technical Adequacy Of Probabilistic Risk Assessment Results For Risk-Informed Activities."

In cases where detailed design information (regarding cable and pipe routing, for example) is not available or when 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 be used to identify vulnerabilities, as well as design and operational requirements, such as ITAAC and COL action items. In

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[4] Risk-informed applications (e.g., implementation of 10 CFR 50.69 or NFPA-806) may involve a scope, level of detail, and/or technical adequacy for the affected areas that is greater than that needed for the COL application.

addition, the bounding assumptions should not mask any risk-significant information about the design and its operation.

C.I.19.5 Technical Adequacy

The quality of the applicant's methodologies, processes, analyses, and personnel associated with the PRA should comply with the provisions for nuclear plant quality assurance (e.g., Appendix B to 10 CFR Part 50). Toward this end, the applicant should adhere to the recommendations provided in Regulatory Guide 1.200 pertaining to quality and technical adequacy. Such adherence will result in a more efficient and consistent NRC staff review process. Alternatively, the applicant should identify, and justify the acceptability of, alternative measures for addressing PRA quality and technical adequacy.

With respect to the use of industry consensus standards and peer reviews to demonstrate PRA technical adequacy, COL applicants should bear in mind the following points:

- As discussed in Section C.I.19.3 of this guide, the scope of the COL applicant's PRA should be a Level 1 and Level 2 PRA that includes internal and external events and addresses all plant operating modes. The possible lack of NRC-endorsed industry consensus standards or peer review processes that address PRA technical elements, initiating events, or plant operating modes within this expected PRA scope does not provide adequate justification for reducing the PRA's scope.

- The scope of a COL applicant's PRA may be somewhat broader than the initial scope of a COL holder's PRA due to the possible lack of NRC-endorsed industry consensus standards and peer review processes. Specifically, 10 CFR Part 50 requires COL holders, by the time of the scheduled fuel load date for the facility, to develop a plant-specific PRA that meets NRC-endorsed consensus standards in effect one year prior to that date. The COL holder's PRA is to be both Level 1 and Level 2, and must include those modes of operation and initiating events for which these standards exist.

- Additional or revised requirements may be needed for advanced LWRs, other reactor designs, or for reactors in the design stage. In this context, 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, including but not limited to digital instrumentation and control system hardware and software, explosive (squib) valves, and the issue of thermal hydraulic (T-H) uncertainties.[5]

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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.

C.I.19.6 Development of Risk Insights

Applicants should use the PRA to develop risk insights about the proposed plant. Consistent with SECY -98-144, the phrase "risk insights" refers to the results and findings that come from risk assessments. Specific to COL applications, major risk insights include information about (1) areas where certain design features are the most effective in reducing risk with respect to operating reactor designs; (2) major contributors to risk, such as hardware failures and human errors; (3) major contributors to maintaining the "built-in" plant safety and ensuring that the risk does not increase unacceptably; (4) major contributors to the uncertainty associated with the risk estimates; and (5) sensitivity of risk estimates to uncertainties associated with failure data, assumptions made in the PRA models, lack of modeling details in certain areas, and previously raised issues.

Designs that have evolved from current plant technology may incorporate features intended to make the plant safer, more available, and easier to operate as compared to currently operating plants. Accordingly, the results of the PRA should indicate that the design represents a reduction in risk compared to existing plants.[2] For this purpose, a broad comparison of risks, by initiating event category, between the proposed design and operating plants (from which the proposed design evolved) can be helpful in identifying the major design features that contribute to the reduced risk of the proposed design compared to operating designs (e.g., passive systems, less reliance on offsite and onsite power for accident mitigation, and divisional separation).

Risk insights should be developed through the use of importance, sensitivity, and uncertainty analyses. Specifically:

- Risk importance studies 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. Such studies should be performed at the system, train, and component level. A variety of importance measures, such as the Fussell-Vesely (F-V) and risk achievement worth (RAW), should be computed.

- Sensitivity studies should be performed to gain insights about the impact of uncertainties (and potential lack of detailed models) on the estimated risk. The objectives of the sensitivity studies are to (1) determine the sensitivity of the estimated risk to potential biases in numerical values, such as initiating event frequencies, failure probabilities, and equipment unavailabilities; (2) determine the impact of potential lack of modeling details on the estimated risk; and (3) determine the sensitivity of the estimated risk to previously raised issues (e.g., motor-operated valve reliability). In addition, for designs using passive safety systems and active "defense-in-depth" systems, sensitivity studies should be performed to investigate the impact of uncertainties on PRA results under the assumption of plant operation without credit for the non-safety-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 non-safety-related systems for regulatory oversight according to the RTNSS process.

- Consistent with Regulatory Guide 1.174, uncertainty studies should consider parametric uncertainties, modeling uncertainties, and completeness uncertainties.

C.I.19.7 PRA Maintenance and Upgrade

The COL applicant should describe its approach for maintaining and periodically upgrading the PRA, as required by 10 CFR Part 50. The PRA maintenance and upgrade program should be based on Regulatory Guide 1.200. Alternatively, the applicant should identify, and justify the acceptability of, alternative measures for maintaining and upgrading the PRA.

The description of the PRA maintenance and upgrade program should address the following:

- Explain how the PRA will be maintained to ensure that it reasonably reflects as-designed, as-to-be-built, and as-to-be-operated conditions. If the applicant uses a screening process that allows some plant modifications to be deferred or not incorporated during the next scheduled PRA maintenance update, the applicant should describe the process and criteria, including documentation requirements. Likewise, if the process includes conditions that require an immediate maintenance update or upgrade of the PRA prior to the next scheduled PRA maintenance update, the applicant should describe the related process and criteria.

- Explain how technical adequacy is determined for pertinent PRA scope areas in 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).

- Describe the use of and criteria for industry peer reviews.

- Discuss the process for dispositioning industry peer review findings.

- Describe 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.

C.I.19.8 Severe Accidents

The applicant should provide a 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, information should be 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.

C.I.19.9 Documentation

The applicant should develop documentation of the PRA model and the analyses performed to support the COL application. This documentation comprises both submittal (i.e., either contained in Chapter 19 of the FSAR or provided in response to a staff request for additional information) and archival (i.e., available for review and audit) documentation.

C.I.19.9.1 *Submittal Documentation*

The applicant should provide an acceptable level of documentation to enable the NRC staff to conclude that (1) the uses and applications identified in Section C.I.19.2 were addressed, (2) the applicant has performed a sufficiently complete and scrutable analysis, and (3) the results support the application for a COL. Consistent with the requirements of 10 CFR Part 52 and the practices for operating plants, the applicant does not need to provide all plant-specific and site-specific PRA information to the NRC.

It is important for applicants to clearly describe the PRA accident sequences in sufficient detail so that the staff can determine their adequacy and reasonableness. One acceptable way of describing the accident sequences is to provide the event tree diagrams used to delineate the accident sequences, along with descriptions of the each event tree heading. This approach provides a convenient and compact method of communicating the sequences and their complex interrelationships, as compared to a lengthy narrative description. Alternatively, the applicant could omit the event tree diagrams in order to minimize the resources required to maintain Chapter 19 of the FSAR; however, this approach may affect the staff's review of the COL application due to the need to conduct audits or develop requests for additional information.

To support the NRC staff's timely review and assessment of the documentation, the applicant should adhere to the recommended format and content identified in Appendix A.

C.I.19.9.2 *Archival Documentation*

The applicant should maintain archival documentation related to the PRA, including a detailed description of engineering analyses conducted and results obtained, irrespective of whether they were quantitative or qualitative or whether the analyses made use of traditional engineering methods or probabilistic approaches. Regulatory Position 1.3 of Regulatory Guide 1.200 provides the attributes and characteristics of archival documentation associated with the COL applicant's PRA. Archival documentation should be maintained as part of the quality assurance program, such that it is available for examination and maintained as lifetime quality records in accordance with Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)."

APPENDIX A

STANDARD FORMAT AND CONTENT FOR FSAR CHAPTER 19

This Appendix provides a typical format and content for Chapter 19 of the FSAR. Use of this format and content guidance will enhance the efficiency and consistency of COL applications and the staff's review of them. 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

- Describe the purpose and objectives of the plant-specific probabilistic risk assessment (PRA) and severe accident evaluations.
- Address the requirements in 10 CFR Part 52 and 10 CFR Part 50, as well as the related Commission policies and positions.
- Address the objectives identified in Section C.I.19.2 of this guide.
- Identify the structure of Chapter 19.

19.1 Probabilistic Risk Assessment

- Provide a description of the PRA and its results.
- Identify the specific PRA information that is docketed (i.e., included in the application), as opposed to information that is retained by the applicant, but available to support NRC reviews and audits.

19.1.1 <u>Uses and Applications of the PRA</u>

19.1.1.1 *Design Phase*

- Describe the use of the PRA in the design phase (through design certification, as appropriate).
- Include FSAR cross-references to specific program descriptions, as appropriate.

19.1.1.2 *COL Application Phase*

19.1.1.2.1 *<u>Use of PRA in Support of Licensee Programs</u>*

- Describe the use of the PRA in the COL application phase, and specifically, its use in support of other licensee programs (e.g., human factors program, severe accident management program).
- Include FSAR cross-references to specific program descriptions, as appropriate.

19.1.1.2.2 *<u>Risk-Informed Applications</u>*

- Identify and describe specific risk-informed applications being implemented during the COL application phase.

- Include FSAR cross-references to specific program descriptions (e.g., 10 CFR 50.69 implementation, NFPA-806 implementation), as appropriate.

19.1.1.3 *Construction Phase*

- Describe the use of the PRA in the construction phase (from issuance of the COL up to initial fuel loading).
- Include FSAR cross-references to specific program descriptions, as appropriate.

Note: This section may need periodic revision to reflect the actual uses and applications of the PRA.

19.1.1.3.1 *Use of PRA in Support of Licensee Programs*

- Describe the use of the PRA in the construction phase to support of other licensee programs (e.g., human factors program).
- Include FSAR cross-references to specific program descriptions, as appropriate.

19.1.1.3.2 *Risk-Informed Applications*

- Identify and describe specific risk-informed applications that will be implemented during the construction phase.
- Include FSAR cross-references to specific program descriptions (e.g., 10 CFR 50.69 implementation, NFPA-806 implementation), as appropriate.

19.1.1.4 *Operational Phase*

- Describe the use of the PRA during plant operations (commencing with initial fuel loading and continuing through plant commercial operation).
- Include FSAR cross-references to specific program descriptions, as appropriate.

Note: This section may need periodic revision to reflect the actual uses and applications of the PRA.

19.1.1.4.1 *Use of PRA in Support of Licensee Programs*

- Describe the use of the PRA during plant operations to support of other licensee programs (e.g., Maintenance Rule, interface with the ROP, reliability assurance program, human factors program, severe accident management program.
- Include FSAR cross-references to specific program descriptions, as appropriate.

19.1.1.4.2 *Risk-Informed Applications*

- Identify and describe specific risk-informed applications that have been implemented during the operational phase.
- Include FSAR cross-references to specific program descriptions (e.g., risk-informed inservice inspection, risk-informed inservice testing, 10 CFR 50.69 implementation, NFPA-806 implementation), as appropriate.

19.1.2 Quality of PRA

- Discuss the quality of the PRA in the context of its uses and the risk-informed applications identified in Section 19.1.1.

19.1.2.1 *PRA Scope*

- Describe the scope of the PRA as discussed in Section C.I.19.3.

19.1.2.2 *PRA Level of Detail*

- Characterize the PRA's level of detail as discussed in Section C.I.19.4.

19.1.2.3 *PRA Technical Adequacy*

- Describe the technical adequacy of the PRA as discussed in Section C.I.19.5.

19.1.2.4 *PRA Maintenance and Upgrade*

- Describe the PRA maintenance and upgrading program as discussed in Section C.I.19.7.

19.1.3 <u>Special Design/Operational Features</u>

- Address the design and operational features intended to improve plant safety, thus reducing risk when compared to currently operating nuclear power plants.

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*

- Identify features and requirements introduced to reduce or eliminate the known weaknesses/vulnerabilities in current reactor designs.
- Indicate the effect of new design features and operational strategies on plant risk.
- Identify PRA-based insights and assumptions used to develop design requirements.

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

- Describe the internal events PRA for operations at power, including its results.
- Organize the information as indicated in Sections 19.1.4.1 and 19.1.4.2 below.

19.1.4.1 *Level 1 Internal Events PRA for Operations at Power*

- Describe the Level 1 internal events PRA for operations at power, including its results.
- Organize the information as indicated in Sections 19.1.4.1.1 and 19.1.4.1.2 below.

19.1.4.1.1 *Description of the Level I PRA for Operations at Power*

- Describe the methodology used to develop the Level 1 PRA model (e.g., fault tree linking, large event tree and small fault tree approach, etc.).
- List the internal initiating events (including internal floods) that are addressed in the PRA.
- List the success criteria used to delineate accident sequences, discuss how they were determined, and identify any T-H codes used.
- Summarize the accident sequences modeled in the PRA.
- List the plant systems and associated functions that are included in the PRA model.
- Identify the source of all numerical data (initiating event frequencies, component failure rates, equipment unavailabilities due to test or maintenance, human error probabilities, common-cause failure parameters, etc.), especially for numerical data that is based on expert judgement or expert elicitation.
- Identify the PRA software platform used to construct the model.
- State the truncation frequency used to solve the PRA model.

19.1.4.1.2 *Results from the Level I PRA for Operations at Power*

- Describe the significant[6] core damage sequences.
- Identify the significant internal initiating events.
- Identify the significant functions, SSCs, and operator actions (typically determined by importance measures such as risk achievement worth and Fussell-Vesely importance measures).
- Identify the PRA assumptions[7] and PRA-based insights.[8]

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[6] 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.

[7] In the context of the PRA, the phrase "assumption" is intended to be consistent with its usage in RG 1.200.

[8] "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

- Discuss the results and insights from importance, sensitivity, and uncertainty analyses.

consistent with its use in referring to the information provided in Table 19.59-29 in the AP600 design control document.

19.1.4.2 *Level 2 Internal Events PRA for Operations at Power*

- Describe the Level 2 internal events PRA for operations at power, including its results.
- Organize the information as indicated in Sections 19.1.4.2.1 and 19.1.4.2.2 below.

19.1.4.2.1 ***Description of the Level 2 PRA for Operations at Power***

- Discuss the interface with the core damage evaluation (Level I PRA).
- Describe the severe accident physical processes/phenomena and modeling.
- List the success criteria used to delineate accident sequences, discuss how they were determined, and identify any T-H codes used.
- Define the accident classes/release categories.
- Characterize the containment ultimate pressure capacity, and explain how it was determined, and identify any computer codes used.

19.1.4.2.2 ***Results from the Level 2 PRA for Operations at Power***

- Describe the significant large release sequences.
- List the significant internal initiating events.
- Identify the significant functions, SSCs, and operator actions.
- Characterize the containment performance.
- Identify the PRA assumptions and PRA-based insights.
- Discuss the results and insights from importance, sensitivity, and uncertainty analyses.

19.1.4.3 *Level 3 Internal Events PRA for Operations at Power (Optional)*

- Describe the Level 3 internal events PRA for operations at power, including its results.
- Organize the information as indicated in Sections 19.1.4.3.1 and 19.1.4.3.2 below.

19.1.4.3.1 ***Description of the Level 3 PRA for Operations at Power (Optional)***

- Discuss the interface with the containment analyses (Level 2 PRA).
- Explain how the fission product source terms were developed, and identify any computer codes used.
- Describe the dose consequence modeling, including evacuation considerations, and identify any computer codes used.
- Describe how inputs to the calculation of offsite consequences were developed (e.g., demography, meteorology).

19.1.4.3.2 ***Results from the Level 3 PRA for Operations at Power (Optional)***

- Describe significant offsite consequence sequences.
- Identify significant functions, SSCs, and operator actions.
- Identify the PRA assumptions and PRA-based insights.
- Discuss the results and insights from importance, sensitivity, and uncertainty analyses.

19.1.5 <u>Safety Insights from the External Events PRA for Operations at Power</u>

- Identify and describe the external events evaluated.
- If some external events were screened out or incorporated into other evaluations, describe the methods used to conduct the screening and bounding analyses.
- Include FSAR cross-references to specific external events, as appropriate.
- Organize the information as indicated in Sections 19.1.5.1 through 19.1.5.N below.

Note: The staff anticipates that seismic events and internal fires will not be screened out or bounded and, therefore, that the applicant will make an evaluation of their risks. Section 19.1.5.1 provides an example format and content for the seismic risk evaluation; similarly, Section 19.1.5.2 provides an example format and content for the internal fire risk evaluation. Other external events (high winds, tornados, external floods, hurricanes, etc.) that are not screened out should be described in Sections 19.1.5.3 through 19.1.5.N (where "N" denotes a consecutive number unique to each external event addressed) using a similar format and content.

19.1.5.1 *Seismic Risk Evaluation*

- Describe the seismic risk evaluation for operations at power, including its results.
- Organize the information as indicated in Sections 19.1.5.1.1 and 19.1.5.1.2 below.

19.1.5.1.1 <u>*Description of the Seismic Risk Evaluation*</u>

- Describe the seismic analysis methodology and approach, including any screening and bounding analyses (e.g., seismic margins analysis).
- Describe the site-specific seismic hazards analysis, and identify the source(s) of information used (e.g., U.S. Geological Survey).
- Describe the SSC fragility analysis, including the use of information about similar components and information developed from expert opinion or expert elicitation.
- Describe the seismic risk accident sequence and system modeling, and identify any computer codes used.

19.1.5.1.2 <u>*Results from the Seismic Risk Evaluation*</u>

- Describe the significant core-damage, large release, and offsite consequence sequences.
- Identify the significant functions, SSCs, and operator actions.
- Identify the PRA assumptions and PRA-based insights.
- Discuss the results and insights from importance, sensitivity, and uncertainty analyses.

19.1.5.2 *Internal Fires Risk Evaluation*

- Describe the internal fire risk evaluation for operations at power, including its results.
- Organize the information as indicated in Sections 19.1.5.2.1 and 19.1.5.2.2 below.

19.1.5.2.1 <u>*Description of the Internal Fire Risk Evaluation*</u>

- Describe the internal fire analysis methodology and approach, including the use of any screening or bounding analyses.
- Explain how the fire initiation frequencies were estimated.
- Describe the propagation of fires, and identify any computer codes used.
- Describe the fire damage modeling, and identify the specific fire-induced failure modes considered in the evaluation.
- Describe the plant response analysis and modeling.

19.1.5.2.2 ***Results from the Internal Fire Risk Evaluation***

- Describe the significant core-damage, large release, and offsite consequence sequences.
- Identify the significant functions, SSCs, and operator actions.
- Identify the PRA assumptions and PRA-based insights.
- Discuss the results and insights from importance, sensitivity, and uncertainty analyses.

19.1.6 Safety Insights from the PRA for Other Modes of Operation

- Describe the risk evaluation for other modes of operation, including its results.
- Organize the information as indicated in Sections 19.1.6.1 and 19.1.6.2 below.

19.1.6.1 ***Description of the Low-Power and Shutdown Operations PRA***

- Identify and describe the other (non-full-power) modes of operation addressed in the risk evaluation.
- If the evaluation of some modes is incorporated into (or bounded by) the evaluations of other modes, describe the methods used to conduct the grouping and bounding analyses.
- Describe the methodology used to develop the low-power and shutdown PRA models.
- List the initiating events (internal and external) that are addressed in the PRA.
- List the success criteria used to delineate accident sequences, discuss how they were determined, and identify any T-H codes used.
- Summarize the accident sequences modeled in the PRA.
- List the plant systems and associated functions that are included in the PRA model.
- Identify the source of all numerical data (initiating event frequencies, component failure rates, equipment unavailabilities due to test or maintenance, human error probabilities, common-cause failure parameters, etc.), especially for numerical data that is based on expert judgement or expert elicitation.
- Identify the PRA software platform used to construct the model.
- State the truncation frequency used to solve the PRA model.

19.1.6.2 ***Results from the Low-Power and Shutdown Operations PRA***

- Describe the significant core-damage, large release, and offsite consequence sequences.
- Identify the significant initiating events, including both internal and external.
- Identify the significant functions, SSCs, and operator actions.
- Identify the PRA assumptions and PRA-based insights.
- Discuss the results and insights from importance, sensitivity, and uncertainty analyses.

19.1.7 <u>PRA-Related Input to Other Programs and Processes</u>

- Describe the specific PRA-related inputs provided to the programs identified in Section 19.1.1.
- Provide cross-references to the specific sections that describe and evaluate each of these programs.

Note: Sections 19.1.1.1 through 19.1.7.5 should discuss the PRA inputs provided to design programs and processes, Maintenance Rule (10 CFR 50.65) implementation, reactor oversight process (MSPI and SDP), the RAP, and the RTNSS program. PRA inputs to other programs and processes should be described in Sections 19.1.7.6 through 19.1.7.N (where "N" denotes a consecutive number unique to each program or process addressed).

19.1.7.1 *PRA Input to Design Programs and Processes*

- Discuss PRA-based insights identified during the design development that ensure the assumptions made in the PRA will remain valid for the as-to-be-built, as-to-be-operated plant.
- Include assumptions regarding SSC and operator performance and reliability, ITAACs, interface requirements; 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 <u>Conclusions and Findings</u>

- Provide a conclusion that the PRA has been used as discussed in Section C.I.19.2.
- Provide a conclusion that the results of the PRA support the decision to issue the COL.

19.2 Severe Accident Evaluation

Describe 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. These features 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 design should 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.

19.2.1 Introduction

- Provide a description of the severe accident evaluation.

19.2.2 Severe Accident Prevention

Provide a deterministic evaluation to show how the plant's severe accident preventive features would cope with the following events:

- Anticipated Transients Without Scram
- Mid-Loop Operations
- Station Blackout
- Fire Protection
- Intersystem Loss of Coolant Accident
- Describe other Severe Accident Preventive Features

19.2.3 Severe Accident Mitigation

- Provide an Overview of the Containment Design
- Describe Severe Accident Progression, both In-and Ex-Vessel
- Describe Severe Accident Mitigation Features for External Reactor Vessel Cooling, Hydrogen Generation and Control, Core Debris Coolability, High-Pressure Melt Ejection, Fuel-Coolant Interactions, Containment Bypass (including Steam Generator Tube Rupture and Intersystem Loss of Coolant Accident), Equipment Survivability, and 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.

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 how the requirement of 10 CFR 50.34(f)(1)(I) has been met.

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 <u>Resolution of Open Items</u>

19.3.2 <u>Resolution of Confirmatory Items</u>

19.3.3 <u>Resolution of COL Action Items</u>