19.0 PROBABILISTIC RISK ASSESSMENT AND SEVERE ACCIDENT EVALUATION

This chapter summarizes information related to the probabilistic risk assessment (PRA) and the severe accident evaluations performed to support design certification of the U.S. EPR. The principal objectives of these analyses are the following:

- To demonstrate that the design poses an acceptably low risk of core damage accidents.
- To identify opportunities for effective and timely improvements during the design phase, through a systematic assessment of the design.
- To provide the foundation for a plant-specific PRA for the combined license (COL) and operating phases.

The COL applicant that references the U.S. EPR design certification will either confirm that the PRA in the design certification bounds the site-specific design information and any design changes or departures, or update the PRA to reflect the site-specific design information and any design changes or departures.

19.0.1 NRC Regulatory Requirements and Related Policies

In performing the PRA and the severe accident assessments, AREVA NP has followed guidance provided by NRC in the following documents:

- NRC Policy Statement, “Severe Reactor Accidents Regarding Future Designs and Existing Plants” (Reference 1).
- NRC Policy Statement, “Safety Goals for the Operations of Nuclear Power Plants” (Reference 2).
- NRC Policy Statement, “Nuclear Power Plant Standardization” (Reference 3).
- NRC Policy Statement, “The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities” (Reference 4).
- SECY-90-016, “Evolutionary Light-Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements” (Reference 5) and the related staff requirements memorandum (SRM) (Reference 6).
- SECY-93-087, “Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor Designs” (Reference 7), and the related SRM (Reference 8).

Reference 1 through Reference 4 above provide guidance regarding the appropriate course of action to address severe accidents and the use of PRA. The SRMs related to SECY-90-016 and SECY-93-087 provide commission-approved guidance for implementing features in new designs to prevent severe accidents and to mitigate their effects, should they occur.

19.0.2 Uses of PRA and Severe Accident Evaluations

The PRA and severe accident evaluations are used to:

- Identify and address potential U.S. EPR design features and plant operational vulnerabilities, where a small number of failures could lead to core damage, containment failure, or large releases.
- Reduce or eliminate the significant risk contributors of existing operating plants that are applicable to the U.S. EPR design by introducing appropriate features and requirements.
- Select among alternative features, operational strategies, and design options.
- Identify risk-informed safety insights based on systematic evaluations of the risk associated with the U.S. EPR design, construction, and operation of the plant such that the following can be identified and described:
  - The U.S. EPR design's robustness, levels of defense-in-depth, and tolerance of severe accidents initiated by internal or external events.
  - The risk significance of specific human errors associated with the U.S. EPR design, including a characterization of the significant human errors that may be used as an input to operator training programs and procedure refinement.
- Demonstrate how the risk associated with the U.S. EPR design compares against the Commission's goals of less than 1E-4/yr for core damage frequency and less than 1E-6/yr for large release frequency. 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.
- Assess the balance of preventive and mitigative features of the U.S. EPR design.
- Demonstrate that the U.S. EPR design represents a reduction in risk compared to existing operating plants.
- Demonstrate that the U.S. EPR 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)).
- 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.

19.0.3 Structure of Chapter 19

The purpose of Chapter 19 is to outline the technical approach used in performing the PRA and the severe accident evaluations and to describe the results and insights obtained from these analyses. The organization and content of Chapter 19 are consistent with regulatory guidance to facilitate review by the NRC. That is, Section 19.1 describes the following elements of the PRA:

- The uses made of the PRA in the design phase. These uses are described in Section 19.1.1.1 and Section 19.1.3.4.
- The steps taken to develop a PRA of sufficient technical quality to support the uses in the design phase and for design certification. Section 19.1.2 addresses the scope and quality of the PRA.
- The methods used in performing the PRA. The methods for the assessment of core damage (i.e., the Level 1 portion) for internal initiating events initiated during power operation are described in Section 19.1.4.1.1. The manner in which the analysis of containment response and large release frequency (the Level 2 portion of the PRA) were conducted is described in Section 19.1.4.2.1. The methods used for the assessment of external events are summarized in Section 19.1.5.1.1 (for PRA-based margins); Section 19.1.5.2.1 (for internal flooding); Section 19.1.5.3.1 (for internal fires); and Section 19.1.5.4 (for other external events). The methods for the assessment of the risk associated with low-power and shutdown operations are discussed in Section 19.1.6.1.
- The results and insights obtained from the PRA. Results for each of the technical areas are also provided in Section 19.1. The results and insights from the assessment of internal events during power operation are presented in Section 19.1.4.1.2. The results from the Level 2 analyses are discussed in Section 19.1.4.2.2. The results from the analyses of various external events are provided in Section 19.1.5.1.2 (for seismic); Section 19.1.5.2.2 (for internal flooding); Section 19.1.5.3.2 (for internal fires); and Section 19.1.5.4 (for other external events). The results related to low-power and shutdown operation are provided inSection 19.1.6.2. The overall conclusions and insights from the PRA are discussed in Section 19.1.8.
- Use of the PRA results as inputs to other programs and processes. These uses are addressed in Section 19.1.1 and Section 19.1.7.

Section 19.2 summarizes the approach taken to address severe-accident issues. These issues are defined in SECY-90-016 (Reference 5) and SECY-93-087 (Reference 7) and in their associated SRMs. The topics addressed in Section 19.2 include the following:

- Measures implemented in the design of the U.S. EPR to prevent severe accidents. These measures, including those intended to address ATWS issues, mid-loop operation, station blackout, fire protection, and interfacing-systems loss of coolant accidents (ISLOCA), are described in Section 19.2.2.

- Design measures to improve the ability to mitigate the consequences of severe accidents. Section 19.2.3 provides an overview of the containment design, describes the in-vessel and ex-vessel progression of core damage events, and addresses specific mitigating features.

- The containment performance capability. The capability of the containment relative to the performance goals identified in SECY-90-016 and SECY-93-087 is described in Section 19.2.4.

- Accident management provisions. Section 19.2.5 addresses guidance developed to aid the operating staff in responding most effectively to the challenges posed by severe accidents. This guidance addresses steps that would be taken to prevent core damage; to terminate core damage such that the core is retained within the reactor vessel; to prevent failure of containment; and to minimize any offsite releases.

- Potential design improvements. The potential improvements to the design that have been considered with respect to reducing the potential for or impact of severe accidents are described in Section 19.2.6.

Finally, Section 19.3 addresses resolution of Open Items, Confirmatory Items, and COL action items identified as unresolved for design certification.

19.0.4 References

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

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

3. NRC Policy Statement, “Nuclear Power Plant Standardization,” Federal Register, 52 FR 34844, September 15, 1987.

4. NRC Policy Statement, “The Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities,” Federal Register, 60 FR 42622, August 16, 1995.

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

6. Staff Requirements Memorandum, U.S. Nuclear Regulatory Commission, June 26, 1990.

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

8. Staff Requirements Memorandum, U.S. Nuclear Regulatory Commission, July 21, 1993.

9. Staff Requirements Memorandum, U.S. Nuclear Regulatory Commission, January 15, 1997.

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