Safety Evaluation Report With Open Items for the U.S. EPR

Portions of Chapter 15, "Transient and Accident Analyses"

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15 TRANSIENT AND ACCIDENT ANALYSES

15.0 Transient and Accident Analyses

AREVA NP (hereafter referred to as AREVA or the applicant) has submitted a Final Safety Analysis Report (FSAR) describing the design, engineering, and other information of the U.S. EPR in connection with its request to the U.S. Nuclear Regulatory Commission (NRC) for design certification. This chapter describes the evaluation of the applicant's FSAR analyses of the U.S. EPR responses to postulated equipment failures or malfunctions. These analyses are used to determine the limiting conditions for operation (LCO), limiting safety system settings (LSSSs), and design specifications for safety-related structures, systems, and components (SSCs).

15.0.1 Radiological Consequence Analysis and Computer Codes Used in Transient and Accident Analyses

This section documents the staff's review of FSAR Tier 2, Sections 15.0.0.1 - 15.0.0.3, 15.0.1, and 15.0.2.

15.0.1.1 *Introduction*

This section addresses FSAR Tier 2, Sections 15.0.0.1, "Classification of Transients and Accidents"; 15.0.0.2, "Accident Analysis Acceptance Criteria"; 15.0.0.3, "Plant Characteristics Considered in the Safety Analysis"; 15.0.1, "Radiological Consequence Analysis"; and 15.0.2, "Computer Codes Used in Analysis"; including the classifications of transients and postulated accidents, accident analysis acceptance criteria, plant characteristics considered in the safety analysis, and computer codes used in the analysis. NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants – LWR Edition," (hereafter referred to as NUREG-0800 or the SRP), Section 15.0.1, "Radiological Consequence Analyses Using Alternative Source Terms," is related to radiological consequence analysis for operating plants and is not applicable to new plants. This topic is covered under FSAR Tier 2, Section 15.0.3, "Radiological Consequences of Design Basis Accidents," for the U.S. EPR.

15.0.1.2 *Summary of Application*

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Sections 2.2.1, "Reactor Coolant System"; 2.2.3, "Safety Injection System and Residual Heat Removal System"; 2.2.4, "Emergency Feedwater System"; and 2.8.2, "Main Steam System."

FSAR Tier 2: The applicant has provided an FSAR Tier 2 system description in Sections 15.0.0.1—15.0.0.3, 15.0.1, and 15.0.2 summarized here, in part, as follows:

FSAR Tier 2, Section 15.0.0.1, "Classification of Transients and Accidents"

The FSAR states that the plant transient and accident analyses provided in Chapter 15 represent a broad spectrum of initiating events that are categorized according to type and frequency. Each initiating event is categorized as an anticipated operational occurrence (AOO), a postulated accident, or a beyond design basis event. AOOs and postulated accidents for the U.S. EPR fall into one of the following event types:

- Radioactive release from a subsystem or component
- Increase in heat removal by the secondary system
- Decrease in heat removal by the secondary system
- Decrease in reactor coolant system (RCS) flow rate
- Reactivity and power distribution anomaly
- Increase in RCS inventory
- Decrease in RCS inventory

The FSAR states that the range of events considered in the U.S. EPR safety analysis is determined by considering potential failures in plant systems or operator errors for each initiating event type. (See FSAR Tier 2, Table 15.0-1, "U.S. EPR Initiating Events.") The categorization according to frequency of occurrence and categorization according to type conform to the guidance in NUREG-0800, Section 15.0, "Introduction – Transient and Accident Analyses."

FSAR Tier 2, Section 15.0.0.2, "Accident Analysis Acceptance Criteria"

FSAR Tier 2, Table 15.0-2, "Accident Analysis Acceptance Criteria," provides the acceptance criteria for AOOs and postulated accidents. These criteria correspond to those in NUREG-0800 and are shown below under Regulatory Basis.

FSAR Tier 2, Section 15.0.0.3, "Plant Characteristics Considered in the Safety Analysis"

The complete operating domain, from power operation to cold shutdown, is considered in the Chapter 15 safety analysis. FSAR Tier 2, Table 15.0-5, "Plant Parameters Used in Accident Analyses," provides the significant plant parameters utilized in the Chapter 15 FSAR analysis for AOOs and postulated accidents. Additional important plant equipment, systems, core reactivity coefficients, power level, control rod parameters, and assumptions used in the safety analysis are provided in FSAR Tier 2, Table 15.0-3, "Plant Operating Modes"; Table 15.0-4, "Nuclear Steam Supply System Power Levels Assumed in the Accident Analysis"; and Tables 15.0-5, "Plant Parameters Used in Accident Analyses," through 15.0-11, "Single Failures Assumed in the Accident Analysis."

FSAR Tier 2, Section 15.0.1, "Radiological Consequence Analysis"

This section of NUREG-0800 is not applicable to new plants. The radiological consequences analyses are addressed in Section 15.0.3 of this report.

FSAR Tier 2, Section 15.0.2, "Computer Codes Used in Analysis"

This FSAR section presents a summary of each principal computer code used in the accident analyses. Additionally, FSAR Tier 2, Table 15.0-6, "Reactivity Coefficients, Core Power, and Computer Codes," lists the code or codes used for each AOO and postulated accident analyses. A summary of each principal computer code used in the accident analyses is presented in the following sections.

CASMO-3G and PRISM

CASMO-3G is an industry standard multi-group, two-dimensional transport theory code used for burnup calculations on pressurized water reactor (PWR) and boiling water reactor (BWR) fuel assemblies and/or simple pin cells. It is used in conjunction with the MICBURN-3 code, which is a multi-group, one-dimensional transmission probability code that is used to calculate microscopic burnup in rods containing absorber (poison) material that is initially homogeneously distributed. The code system is designed to handle a geometry consisting of cylindrical fuel rods of varying compositions in a square pitch array with allowance for fuel rods loaded with a burnable absorber, burnable absorber rods, rod cluster control rods, in-core instrument channels, water gaps, boron steel curtains, and cruciform control rods in the regions separating fuel assemblies. The code system is used to generate assembly lattice cross section data as a function of burnup, discontinuity factors, and heterogeneous form factors, which are then passed to the core simulator code (PRISM).

PRISM is a three-dimensional core simulator code used to calculate core power distributions and perform reactivity analyses using assembly data from the CASMO-3G code. The code uses a nodal expansion method to solve the two-group neutron diffusion theory representation of the reactor core. It provides for thermal-hydraulic feedback, modeling of equilibrium or time-dependent xenon and samarium, and isotopic depletion. In addition, it allows for the generation of pin-by-pin power distributions using a pin power reconstruction technique. A continuous cross section representation was developed covering all combinations of thermal hydraulic parameters possible in stationary reactor states.

ORIGEN 2.1

ORIGEN 2.1 is a computer code for calculating the buildup, decay, and processing of radioactive materials. ORIGEN 2 generates isotope atom densities as a function of time used to generate macroscopic cross-sections. ORIGEN 2.1 includes additional libraries for standard and extended-burnup PWR and BWR calculations, which are documented in Oak Ridge National Laboratory (ORNL)/Technical Manual (TM)-11018. ORIGEN 2 was developed by Oak Ridge National Laboratory, has been extensively benchmarked, and is used throughout the nuclear industry. The code is described in ORNL-5621, “ORIGEN2-A Revised and Updated Version of the Isotope Generation and Depletion Code,” July 1980.

NEMO-K

NEMO-K is a three-dimensional, reactor kinetics code incorporating time-dependent solutions for neutronics, fuel temperature, and coolant properties into the steady-state NEMO computer code. NEMO-K solves the nodal balance equations in three dimensions to determine the neutron flux, source, relative power density (including pin power reconstruction), and reactor core reactivity. The code tracks both the power generated within the fuel and that deposited directly in the water. The fuel pin temperature model assumes a circular rod with azimuthally symmetric heat generation and no axial conduction. The transient thermal-hydraulic model assumes no mixing between channels and a constant pressure throughout the core.

LYNXT

LYNXT is the primary code used to perform thermal-hydraulic licensing calculations. Its primary role is determining transient departure from nucleate boiling ratio (DNBR). LYNXT is a versatile thermal-hydraulics cross-flow code capable of predicting flow and temperature (enthalpy)

distributions in confined geometries where wall shear forces are more dominant than intra-fluid shear forces.

LYNXT is an improved version of the COBRA IV-1 code developed at Battelle Pacific Northwest Laboratories under the sponsorship of the Energy Research and Development Administration and the NRC.

Fuel Rod Initial Condition Codes

In FSAR Tier 2, Section 15.0.0.3.3, "Reactivity Coefficients Assumed in the Accident Analysis," the applicant notes that three additional codes are used to establish the initial conditions for the fuel rod models in S-RELAP5. RODEX2 is used to compute the fuel initial conditions for the small-break loss-of-coolant accident (SBLOCA) analysis. RODEX3 is used to compute the fuel initial conditions for the realistic large-break loss-of-coolant accident (RLBLOCA) analysis. The initial fuel conditions for all transient analysis that are not LOCA analyses are based upon COPERNIC2.

S-RELAP5 Application for Non-LOCA Events

S-RELAP5 code is the principal tool for U.S. EPR non-loss-of-coolant accident (non-LOCA) analyses. S-RELAP5 is a general purpose thermal-hydraulic transient simulation code that evolved from the RELAP5 family of computer codes developed originally by the Idaho National Engineering Laboratory (INEL) for the NRC. The S-RELAP5 code is capable of simulating the hydraulic and thermal phenomena necessary to predict transients in both nuclear and non-nuclear systems involving mixtures of steam, water, noncondensable gas, and solute. S-RELAP5 includes hydrodynamic models, heat transfer and heat conduction models, a fuel model, a reactor kinetics model, and control system and trip system models.

The S-RELAP5 code includes many generic component models from which general systems can be simulated. These components include pumps, valves, pipes, heat releasing or absorbing structures, reactor point kinetics, electric heaters, turbines, separators, accumulators, and control and trip system components. Special process models are included for effects such as form loss, flow at an abrupt area change, branching, choked flow, counter-current flow limit (CCFL), boron tracking, and noncondensable gas transport. The reactor fuel is represented as a heat releasing or absorbing structure.

S-RELAP5 Application for Small-Break LOCA Events

S-RELAP5 is used for SBLOCA calculations with modeling consistent with Appendix K, "ECCS Evaluation Models," to Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities."

S-RELAP5 Application for Large Break LOCA Events

S-RELAP5 is the principle code used in Realististic Large Break LOCA (RLBLOCA) methodology which is based upon nonparametric statistics. S-RELAP5 modeling is consistent with 10 CFR 50.46.

ITAAC: The inspection, test, analysis, and acceptance criteria (ITAAC) associated with FSAR Tier 2, Section 15.0.1 and 15.0.2 are given in FSAR Tier 1, Section 2.2.1 in Table 2.2.1-5, "RCS ITAAC," Item 7.6; Section 2.2.3 in Table 2.2.3-1, "SIS/RHRS Equipment

Mechanical Design," Item 7.5; Section 2.2.4 in Table 2.2.4-3, "EFWS ITAAC," Item 7.2; and Section 2.8.2 in Table 2.8.2-3, "MSS ITAAC," Items 7.2 through 7.5.

Technical Specifications: There are no Technical Specifications for this area of review.

15.0.1.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Sections 15.0, "Introduction - Transient and Accident Analyses," and 15.0.2, "Review of Transient and Accident Analysis Method," and are summarized below. Review interfaces with other Standard Review Plan (SRP) sections also can be found in NUREG-0800, Sections 15.0 and 15.0.2.

1. 10 CFR Part 20, "Standards for Protection Against Radiation"

2. 10 CFR Part 50 (especially 10 CFR 50.46 and Appendix A)

3. 10 CFR Part 52, "Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants"

4. 10 CFR Part 100, "Reactor Site Criteria"

The following General Design Criterion (GDC) from 10 CFR Part 50, Appendix A are relevant to SRP Section 15.0:

1. GDC 2, "Design Bases for Protection Against Natural Phenomena," as it relates to the seismic design of structures, systems, and components (SSCs) whose failure could cause an unacceptable reduction in the capability of the residual heat removal system.

2. GDC 4, "Environmental and Dynamic Effects Design Bases," as it relates to the requirement that SSCs important to safety be designed to accommodate the effects of and be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accident conditions, including such effects as pipe whip and jet impingement.

3. GDC 5, "Sharing of Structures, Systems, and Components," as it relates to the requirement that any sharing among nuclear power units of SSCs important to safety will not significantly impair their safety function.

4. GDC 10, "Reactor Design," as it relates to the RCS being designed with appropriate margin to ensure that specified acceptable fuel design limits are not exceeded during normal operations, including AOOs.

5. GDC 13, "Instrumentation and Control," as it relates to instrumentation and controls provided to monitor variables over anticipated ranges for normal operations, for AOOs, and for accident conditions.

6. GDC 15, "Reactor Coolant System Design," as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations, including AOOs.

7. GDC 17, “Electric Power Systems,” as it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of SSCs important to safety. The safety function for each system (assuming the other system is not working) shall be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded during an AOO and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.

8. GDC 19, “Control Room,” as it relates to the requirement that a control room be provided from which personnel can operate the nuclear power unit during both normal operating and accident conditions, including a LOCA.

9. GDC 20, “Protection System Functions,” as it relates to the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that the plant does not exceed specified acceptable fuel design limits during any condition of normal operation, including AOOs.

10. GDC 25, “Protection System Requirements for Reactivity Control Malfunctions,” as it relates to the requirement that the reactor protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods.

11. GDC 26, “Reactivity Control System Redundancy and Capability,” as it relates to the reliable control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded even during AOOs. This is accomplished by ensuring that the applicant has allowed an appropriate margin for malfunctions such as stuck rods.

12. GDC 27, “Combined Reactivity Control Systems Capability,” and GDC 28, “Reactivity Limits,” as they relate to the RCS being designed with an appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.

13. GDC 29, “Protection Against Anticipated Operational Occurrences,” as it relates to the design of the protection and reactivity control systems and their performance (i.e., to accomplish their intended safety functions) during AOOs.

14. GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a nonbrittle manner and that the probability of propagating fracture is minimized.

15. GDC 34, “Residual Heat Removal,” as it relates to the capability to transfer decay heat and other residual heat from the reactor so that fuel and pressure boundary design limits are not exceeded.

16. GDC 35, “Emergency Core Cooling,” as it relates to the RCS and associated auxiliaries being designed to provide abundant emergency core cooling.

17. GDC 55, “Reactor Coolant Pressure Boundary Penetrating Containment,” as it relates to the isolation requirements of small-diameter lines connected to the primary system.

18. GDC 60, “Control of Releases of Radioactive Materials to the Environment,” as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment.

19. GDC 61, “Fuel Storage and Handling and Radioactivity Control,” as it relates to the requirement that the fuel storage and handling, radioactive waste, and other systems that may contain radioactivity be designed to ensure adequate safety under normal and postulated accident conditions.

Acceptance criteria adequate to meet the above requirements include:

1. For AOOs:

    - Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code.

    - Fuel cladding integrity shall be maintained by ensuring that the minimum departure from nucleate boiling ratio (MDNBR) remains above the 95 percent probability, with 95 percent confidence (95/95) DNBR limit for PWRs.

    - An AOO should not generate a postulated accident without other faults occurring independently or result in a consequential loss of function of the RCS or reactor containment barriers.

2. For postulated accidents:

    - Pressure in the RCS and main steam system should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

    - Fuel cladding integrity will be maintained if the MDNBR remains above the 95/95 DNBR limit for PWRs.

    - The release of radioactive material shall not result in offsite doses in excess of the guidelines of 10 CFR Part 100.

    - A postulated accident shall not, by itself, cause a consequential loss of required functions of systems needed to cope with the fault, including those of the RCS and the reactor containment system.

3. For LOCAs:

    - The calculated maximum fuel element cladding temperature shall not exceed 1,200 °C (2,200 °F).

    - The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation.

    - The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders

surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

- Calculated changes in core geometry shall be such that the core remains amenable to cooling.
- After any calculated successful initial operation of the emergency core cooling system (ECCS), the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

15.0.1.4 *Technical Evaluation*

Categorization and Classification of Events

The staff reviewed the categorization and classification of events that are analyzed along with the corresponding acceptance criteria. The categorization and classification of events corresponds to the guidance in Regulatory Guide (RG) 1.206, "Combined License Applications for Nuclear Power Plants (LWR Edition)," June, 2007, and SRP Section 15.0, "Introduction-Transient and Accident Analyses," and, therefore, ensures that a broad spectrum of events has been considered and that the events are appropriately classified as anticipated operational occurrences or accidents with the proper associated acceptance criteria.

Computer Codes Used

The application of computer codes and simulation models to the U.S. EPR are described in FSAR Tier 2, Section 15.0.2. AREVA submitted ANP-10263P, "Codes and Methods Applicability Report of the U.S. EPR," to explain the applicability of the computer codes listed in FSAR Tier 2, Section 15.0.2. The staff's Final Safety Evaluation Report (FSER) for Topical Report (TR) ANP-10263P, "Codes and Methods Applicability Report for the U.S. Evolutionary Power Reactor (U.S. EPR)", August 8, 2007, approved the application of computer codes and simulation models described in ANP-10263P to the U.S. EPR.

The approval was based on the design of the U.S. EPR, and prior past approvals of codes summarized below:

The staff has previously reviewed and accepted the CASMO3 and PRISM codes, in an October 29, 1996, letter to Mr. H. D. Curet, Siemens Power Corporation, "Acceptance for Referencing of TR EMF-96-029(P) Vols. 1 and 2, Reactor Analysis System for PWR'S."

The staff has previously reviewed and accepted the NEMO-K code, in a June 10, 1998, letter to Mr. F. McPhatter, Framatome Technologies, "Acceptance for Referencing of Licensing TR BAW-10221 P, NEMO-K, A Kinetics Solution in NEMO."

AREVA report ANP-10286P, Revision 0, "U.S. EPR Rod Ejection Accident Methodology Topical Report," November 2007, describes the application of NEMO-K as part of a code suite used to analyze the ejected rod accident for the U.S. EPR. The applicability of this code suite is reviewed in the safety evaluation report for ANP-10286P.

The staff has previously reviewed and accepted the LYNXT code, in a December 3, 1985, letter to Mr. J. H. Taylor, Babcock & Wilcox, "Acceptance for Referencing of Licensing Topical Report

BAW-10156, 'LYNXT- Core Transient Thermal- Hydraulic Program'." The staff also reviewed and accepted Revision 1 of the LYNXT code, TR BAW-10156A, Revision 1, August 1993.

The staff previously approved RODEX2, Fuel Rod Thermal-Mechanical Evaluation Model, August 1986, and RODEX 3, ANF-90-145-(P)(A) RODEX3 Fuel Rod Thermal-Mechanical Response Evaluation Model, Volume 1, "Theoretical Manual," and Volume 2, "Thermal and Gas Release," April 1996. The application of RODEX2 and RODEX 3 to the U.S. EPR is further discussed in Section 15.6 of this report.

In the FSER for AREVA TR ANP-10263P-A, Revision 0, "Codes and Methods Applicability Report for the U.S. Evolutionary Power Reactor," the staff determined there is reasonable assurance that the response of the U.S. EPR to non-LOCA transient and accidents can be evaluated with the currently approved S-RELAP5 code.

The staff has previously reviewed and accepted the S-RELAP5 code for Appendix K-based SBLOCA evaluations in a March 15, 2001, letter to Mr. James F. Mallay, Framatome ANP, "Acceptance for Referencing of Licensing Topical Report EMF-2328P, Revision 0, PWR Small Break LOCA Evaluation Model, S-RELAP5 Based." (Reference AREVA, "PWR Small-Break LOCA Evaluation Model, S-RELAP5 Based," EMF-2328 (P)(A), Revision 0, March 2001.

In the FSER for AREVA TR ANP-10263P-A, the staff review centered on U.S. EPR design features which might require new models or methods for licensing analyses.

The U.S. EPR does not have traditional high head safety injection pumps as part of the ECCS. The U.S. EPR relies on safety-related steam generator (SG) cooling to bring the primary system pressure down to the shut-off head of the medium head safety injection pump. The main steam relief train (MSRT) is also used to manage a SG tube rupture (SGTR) event. Therefore, the modeling of the SG, including reflux condensation and natural circulation, was the focus of the staff's review for the SBLOCA. The staff determined these models acceptable.

The applicant committed to performing studies on the modeling of the U-tubes in the Appendix K SBLOCA evaluation model to justify the licensing model. The results of these sensitivity studies are described in Section 15.6.2 of this report. The staff determined this approach acceptable for justifying the U.S. EPR SG model development for use in the Appendix K SBLOCA evaluation model.

The staff has previously reviewed and approved AREVA Realistic Large Break LOCA Evaluation Model, EMF-2103(P)(A), April 2003, for operating reactors. In a March 2007, letter, AREVA submitted TR ANP-10278P, Revision 0, "U.S. EPR Realistic Large Break Loss of Coolant Accident Topical Report," for staff review and approval. The review of this topical report is described in Section 15.6.5.1, "Large Break Loss of Coolant Accident," of this report.

Acceptance Criteria

FSAR Tier 2, Table 15.0-2, "Accident Analysis Acceptance Criteria," shows the acceptance criteria for AOOs and accidents. The staff notes that the criteria are identical to the acceptance criteria in SRP Section 15.0, and, therefore, are acceptable.

The staff issued requests for additional information (RAIs) applicable to the transient and accident analysis concerning the initial conditions, time in cycle, and postulated single failures, which affect the course of each analyzed event. Conditions that cause one criteria to reach a

maximum or minimum value may not result in another criteria reaching a limiting value. Responses to these RAIs, which are applicable to all transients and accidents, are described in the following paragraphs.

The U.S. EPR FSAR did not clearly identify for each acceptance criteria the parameters and assumptions used in the analysis of each transient and accident. Only the limiting analysis is provided. In RAI 34, Question 15-1, the staff requested that the applicant provide the assessment for the parameters, initial conditions, and single failures used in various transients and accidents to support the values and sequence of events assumed in each event that would lead to the most conservative results with respect to each acceptance criteria. In an October 31, 2008, response to RAI 34, Question 15-1, the applicant proposed expanded tables and modified the FSAR to describe the limiting cases for the acceptance criteria for each event. For each transient and accident, the applicant provided a revised Table of Plant Parameters Used in Accident Analyses; an expanded Table of Reactivity Coefficients, Scram Reactivity, and Computer Codes; a revised Table of Single Failures Assumed in the Accident Analysis; and a new Table of Transient Analysis Limiting Cases. The revised and new tables are contained in Revision 1 of the FSAR. The staff finds that the revised tables provide the requested information, and are therefore acceptable. Therefore, the staff considers RAI 34, Question 15-1 resolved. In RAI 34, Question 15-2, the staff requested that the applicant confirm that each of the transients and accidents analyzed has been assessed against multiple acceptance criteria including specified acceptable fuel design limits (SAFDL) including DNBR, maximum primary pressure, and maximum secondary side pressure, and MDNBR. In an October 31, 2008, response to RAI 34, Question 15-2, the applicant stated that, in general, one criterion is dominant for a given transient or accident. In those cases where more than one criterion could be challenged, each applicable criterion is specifically analyzed. The applicant modified the FSAR, and the staff confirmed that in Revision 1 of the FSAR, the applicant added a tabulation of which criteria are addressed for each transient and accident, and the resulting limiting case and justification. Therefore, the staff considers RAI 34, Question 15-2 resolved.

Associated with the above RAIs, in RAI 34, Question 15-3, the staff also requested that the applicant extend the tables shown in FSAR Tier 2, Chapter 15.0, to show for each transient and accident the limiting power, temperatures, flows, levels, scram reactivity, reactivity coefficients, heat transfer coefficients, and degree of SG tube plugging. The applicant agreed to modify existing tables and add an additional table to provide the information. In an October 31, 2008, response to RAI 34, Question 15-3, the applicant provided revisions to the tables contained in FSAR Tier 2, Section 15.0 to address the staff's concern. The staff reviewed the revisions and found them to be acceptable. The staff confirmed that the revisions were contained in Revision 1 of the FSAR that was submitted on May 29, 2009. Therefore, the staff considers RAI 34, Question 15-3 resolved.

Limiting Single Failure

In FSAR Tier 2, Section 15.0.0.3.8 "Limiting Single Failure," the applicant states that:

The accident analyses presented in Chapter 15 incorporate the most limiting active single failure of a safety-related system. FSAR Tier 2, Table 15.0-11 gives the most limiting single failure for each event. Passive failures are not considered, except as event initiators, during the first 24 hours of the event.

The staff notes that with a four division protective system, and four divisions of 100 percent safety systems each powered by an emergency diesel generator, the identification of the limiting

single failure is not straightforward. To ensure that the limiting single failure had been identified, in RAI 34, Question 15-4, the staff requested that the applicant provide in the analysis section of each transient and accident the technical bases to demonstrate that the limiting single failure has been selected and the physical bases identified. The applicant was requested to show that the limiting single failure is selected for each acceptance criteria. In an October 31, 2008, response to RAI 34, Question 15-4, the applicant agreed to modify the FSAR to include a discussion of the technical basis in order to demonstrate that the limiting single failure is selected. The information is contained in Revision 1 of the FSAR that was submitted on May 29, 2009. The staff reviewed the justification for each identified single failure in each postulated transient and accident shown in FSAR Tier 2, Table 15.0.11, "Single Failures Assumed in the Accident Analysis," and based on the staff's understanding of the system dynamics concurs in the identification of single failures. Therefore, the staff considers RAI 34, Question 15-4 resolved.

Loss of Offsite Power

The FSAR states that both loss of offsite power (LOOP) and offsite power-available conditions are considered for each event. LOOP is not considered the limiting single failure and an additional failure is postulated in the transient and accident analysis. An audit of AREVA's calculations confirmed this statement. However, only the worst condition is shown in the FSAR. The staff's review of the applicant's response to RAI 34, Question 15-4 confirmed that the worst condition has been analyzed and presented in the FSAR.

Limiting Initial Conditions

In RAI 34, Question 15-5, the staff requested that the applicant provide information to demonstrate that the limiting analysis conditions are selected. Specifically, the applicant was requested to state the physical or phenomenological bases with respect to: 1) Why retention or LOOP is limiting; 2) why beginning or end of cycle is limiting; and 3) why hot full power (HFP) or hot zero power (HZP) or an intermediate power is limiting. The staff requested that the applicant consider each acceptance criteria and show that limiting conditions are selected for consideration of each criterion (e.g., MDNBR, peak RCS pressure, and peak secondary side pressure) may differ with limiting analysis conditions. In an October 31, 2008, response to RAI 34, Question 15-5, the applicant stated that the FSAR would be modified to include a discussion of the process for selection and the technical bases for the values supporting the limiting cases. The staff reviewed the proposed changes and found them to be acceptable. The staff confirmed that Revision 1 of the FSAR, submitted on May 29, 2009, contained the proposed changes to the FSAR and, therefore, consider RAI 34, Question 15-5 resolved.

Models

A review of the transient and accident analysis shows that some transients have been evaluated using a single channel core while other transients have used a multiple core channel model. In RAI 34, Question 15-6, the staff requested that the applicant provide a description of the S-RELAP5 model used in non-LOCA asymmetric calculations. In an October 31, 2008, response to RAI 34, Question 15-6, the applicant stated that the FSAR Tier 2, Chapter 15 transient and accident analysis used the following five different S-RELAP5 models depending on the event modeled and whether the event response was symmetrical.

- Non-loss-of-coolant accident
- Main steamline break (MSLB) split core
- Non-LOCA split core
- Small-break loss-of-coolant accident
- Large-break loss-of-coolant accident

The non-LOCA S-RELAP5 model is used for the majority of the non-LOCA transients and accidents where the nuclear steam supply system (NSSS) response is symmetrical. The MSLB split core model is used for MSLB events where the broken loop cold leg feeds a separate sector of the core. These models have been previously reviewed and approved by the staff in the FSER on TR ANP-10263P, "Codes and Methods Applicability Report for the U.S. Evolutionary Power Reactor (U.S. EPR)," August 8, 2007.

The non-LOCA split core model is used for non-LOCA events whose response is asymmetric. The model used is a variation of the MSLB and non-LOCA models, and is used for the feedwater flow increase event. The non-LOCA core model is modified by partitioning the single channel core into a two-channel arrangement with no cross-flow or mixing. It is similar to the MSLB model.

The SBLOCA and LBLOCA each have their own unique model. The SBLOCA model is described in AREVA TR ANP-10263P-A and EMF-2328P-A. The LBLOCA model is described in EMF-2103P-A.

The staff finds the applicant's response to RAI 34, Question 15-6 acceptable. The model for non-LOCA transients where the system response is symmetric, such as secondary side upset events, and the model for the asymmetric steamline break analysis have been previously approved. Similarly, the models for SBLOCA and LBLOCA have been previously approved. The non-LOCA split core model used in the feedwater flow increase event is a straightforward variant of previously approved models. The staff finds that it is an acceptable modification of previously approved models because the fundamental conservation equations, constitutive data and associated flow regimes are unchanged, and the split-core nodalization does not introduce new phenomenology. The feedwater flow increase is further discussed below in Section 15.1.1.2.2 of this report. Therefore, the staff considers RAI 34, Question 15-6 resolved.

The decay heat model used in the transient and non-LOCA accident analyses is not specifically stated in the FSAR. The staff requested this information in RAI 34, Question 15-7. In an October 31, 2008, response to RAI 34, Question 15-7, the applicant stated that the decay heat curve used to calculate the total core thermal power varies between the LOCA events and the non-LOCA events. For the SBLOCA, the 1973 American Nuclear Society (ANS) Standard, plus 20 percent is used. For the RLBLOCA, the ANS 1979 Standard is used as described in ANP-10278P. The non-LOCA transients utilize the 1973 ANS Standard with a multiplier of one.

The staff confirmed that the SBLOCA analysis which uses the 1973 Standard, plus 20 percent, bounds the requirements of 10 CFR Part 50, Appendix K, "ECCS Evaluation Models." The RLBLOCA analysis uses the 1979 ANS Standard, which conforms to the guidance contained in RG 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance." Unlike the ECCS analysis, there is no requirement for non-LOCA events to use a decay heat

multiplier greater than one. Furthermore, the 1973 ANS Standard is known to be conservative relative to more modern versions of the ANS decay heat standard. The staff finds the applicant's response to RAI 34, Question 15-7 acceptable, since it complies with the regulatory requirements and conforms to the regulatory guidance identified above. Therefore, the staff considers RAI 34, Question 15-7 resolved.

In RAI 147, Question 15.06.05-41, the staff requested that the applicant specify what parameters are transferred from COPERNIC to S-RELAP5 and to explain how burnup effects and their uncertainties are taken into consideration. In a February 19, 2009, response to RAI 147, Question 15.06.05-41, the applicant stated that COPERNIC provides the fuel thermal conductivity, heat capacity, and gap conductance for S-RELAP5 at any specified burnup level. Parameters are selected to provide a conservative response for the transient under consideration. For example, during a transient, the reactivity contribution from moderator and fuel temperature changes are influenced by the gap conductance. For transients, such as rod withdrawals, where the fuel temperature feedback limits the peak power, a conservatively high gap conductance is used to minimize the Doppler feedback. A high gap conductance is also conservative for DNBR calculations, because it maximizes the cladding heat flux. The non-LOCA transient analysis uses a high gap conductance for all cases except loss of flow events, since it results in higher fuel temperatures and clad heat flux later in the event when the core flow has decreased. Reduced flow, in conjunction with high clad heat flux, minimizes DNBR. Sensitivity studies have shown that a low gap conductance is conservative for these events. The staff finds the applicant's response acceptable and, therefore, considers RAI 147, Question 15.06.05-41 resolved.

Setpoints

Departure from nucleate boiling and linear power density (LPD) are computed online. AREVA report ANP-10287P, Revision 0, "In-core Trip Setpoint and Transient Methodology for U.S. EPR," November 2007, documents the analytical methodology used to determine the setpoints for the in-core-based DNB and LPD LSSS, limitation, and LCO functions in the U.S. EPR. The DNB setpoints are established such that the DNB-limiting pin in the core will not experience DNB during DNB basis events, at 95 percent probability, with 95 percent confidence. The LPD setpoints are established, such that the location in the core with peak LPD will not exceed either fuel centerline melt (FCM) or clad strain limits during LPD basis events, at 95 percent probability and with 95 percent confidence. ANP-10287P has been approved by the staff with conditions (Reference U.S. NRC, Safety Evaluation for AREVA TR ANP-10287P, "Incore Trip Setpoint and Transient Methodology for U.S. Evolutionary Power Reactor," June 16, 2009). The methodology is not applicable to mixed fuel cores and confirmatory analysis for the first and reload cores will be required since actual uncertainties and setpoint values are not currently available. While this topical report describes the setpoint methodology, it does not provide the actual setpoints and uncertainties. The uncertainty values used in the topical report are considered to be representative. The actual values will be confirmed before implementation through the ITAAC process.

FSAR Tier 2, Section 15.0 does not explain how the system code interacts with the setpoint analysis. In RAI 34, Question 15-8, the staff requested that the applicant explain how the system code interacts with the computer code, which calculates the DNBR trip. In a proprietary response to RAI 34, Question 15-8, dated September 24, 2008, the applicant explained how the system interacts with the setpoint analysis. The applicant's response clarifies the interface

between the system and computer code. Therefore, the staff considers RAI 34, Question 15-8 resolved.

Non-Safety-Related Systems Assumed in the Analysis

The plant systems and components that mitigate postulated events in the accident analyses are provided in FSAR Tier 2, Table 15.0-10, "Plant Systems Used in the Accident Analysis." Safety-related systems are credited to mitigate events in the design-basis accident (DBA) analyses for the U.S. EPR.

FSAR Tier 2, Section 15.0.0.3.6 states that non-safety-related systems, including control systems such as pressurizer (PZR) spray control and automatic average coolant temperature (ACT) control, are simulated when their operation makes the response of the event more severe. In this case, it is assumed that they function as designed. Control systems are assumed to fail as is when this makes the event more severe. Failures of other non-safety-related systems are considered only as event initiators. The staff finds that these input assumptions make the event analysis results more severe and, therefore, are conservative.

Similarly, FSAR Tier 2, Section 15.0 states that a turbine trip (TT) is generated on a reactor trip (RT). This signal closes the turbine control and stop valves, terminating steam flow to the turbine. Crediting these non-safety-related backup protective systems and components in the DBA analysis following a RT follows the regulatory position stated in NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director NRR to NRR Staff." This is consistent with past staff practices dating to 1976. Therefore, the staff finds this approach acceptable.

Operator Actions

FSAR Tier 2, Section 15.0.0.3.7 states that operator action is credited in certain analyses to mitigate postulated events. Operator actions are credited for long term mitigation of a feedwater line break, MSLB, SGTR, ECCS switchover to hot leg injection and extra borating system (EBS) malfunction, as well as bringing the plant to residual heat removal (RHR) entry conditions or establishing long term cooling following a LOCA. In such cases, the action is not credited in the analysis before 30 minutes after event initiation if the action can be performed from the main control room (MCR), and 60 minutes if it cannot be performed from the MCR. The staff finds that this conforms to the staff guidance in NUREG-0700, "Human-System Interface Design Review Guidelines," Revision 2, May 2002, and is therefore acceptable.

Post Chapter 15 Event Cooldown is addressed later as part of long term cooling in Section 15.6, "Decrease in Reactor Coolant Inventory Events," of this report.

Three Mile Island and Generic Issues

RG 1.206, "Combined License Applications of Nuclear Power Plants," June 2007, Section C.I.15, "Transient and Accident Analyses," provides guidance to combined license (COL) applicants to address Three Mile Island (TMI) and Generic Issues and Bulletins and provides a list of items to be addressed. FSAR Tier 2, Tables 15.0-57, "TMI Action Plan Items" through 15.0-61, "NRC Generic Letters," present the disposition of the NRC issues given in RG 1.206, Section C.I.15. The applicant has systematically addressed these issues and provided a cross-reference to where resolution of each issue is described in the FSAR. A

complete list of TMI and Generic Issues is found in FSAR Tier 2, Section 1.9, "Conformance with Regulatory Criteria." See Section 1.9 of this report for additional details.

Emergency Operating Procedures

FSAR Tier 2, Section 13.5 indicates that a COL applicant who references the U. S. EPR design certification will provide site-specific information for Emergency Operating Procedures (EOPs). The staff has reviewed FSAR Tier 2, Section 13.5.2.1 regarding the EOPs and finds that the description provided in FSAR Tier 2, Section 13.5 does not provide guidance to COL applicants to perform the necessary verification and validation of the EOPs that would assure the site-specific EOPs are consistent with the safety analyses assumptions contained in FSAR Tier 2, Chapter 15. In RAI 415, Question 15-9, the staff requested that the applicant propose a COL action item to require that the COL applicant perform verification and validation of its site-specific EOPs using plant simulators to assure that the operator action time assumed in the safety analyses documented in FSAR Tier 2, Chapter 15 are achievable to assure that the safety analyses of record remains valid. The staff considers this issue an open item in the staff review and will remain open until a satisfactory resolution is reached. **RAI 415, Question 15-9, which is associated with the above request, is being tracked as an open item**.

Technical Specifications

In RAI 271, Question 16-293, the staff requested that the applicant provide an assessment to confirm that all the LCO values in the proposed technical specifications (TS) are consistent with the initial conditions assumed in the safety analyses. In particular, the staff cited the proposed pressurizer minimum water level of 75 percent specified in LCO 3.4.9.a as an example where a TS requirement is not consistent with the initial condition assumed in the safety analyses (e.g., a pressurizer water level of 59 percent is assumed in heat-up transients in FSAR Tier 2, Section 15.2). In a September 30, 2009, response to RAI 271, Question 16-293, the applicant proposed to revise LCO 3.4.9.a and the associated discussion in the TS Bases B 3.4.9 to reflect the assumed value of 59 percent in a feedwater line break event which is identified as the limiting event for pressurizer water level concerns. The staff finds the revised LCO 3.4.9.a and its associated TS Bases B 3.4.9 acceptable.

However, the applicant has not provided an evaluation of all the proposed TS values in FSAR Tier 2, Chapter 16 to confirm that all assumptions in FSAR Tier 2, Chapter 15 safety analyses are within the limiting conditions for operation specified in the TS. The information is needed to confirm the staff conclusion that all assumptions of the safety analysis are met. In a follow-up RAI 311, Question 16-317, the staff requested that the applicant provide an overall assessment as stated in the original RAI 271, Question 16-293. **RAI 311, Question 16-317 is being tracked as an open item**.

Main Feedwater Flow Measurement Uncerceties Affecting Safety Analyses Assumptions

FSAR Tier 2, Section 15.0.0.3.1, "Design Plant Conditions and Initial Conditions", indicated that a heat balance measurement uncertainty of plus or minus 22 MWt is applicable to the rated core thermal power of 4590 MWt, which represents an uncertainty of approximately 0.48 percent for the maximum core power assumed in the accident analyses. The applicant states that the core power is determined using a secondary-side heat balance, and that the relatively low heat balance uncertainty is achieved by using an ultrasonic flow meter for the feedwater flow rate. The maximum power level assumed in the accident analyses are described in FSAR Tier 2, Table 15.0.4, "Nuclear Steam Supply System Power Levels Assumed in the Accident Analyses"

and FSAR Tier 2, Table 15.0.5, ”Plant Parameters Used in Accident Analyses” lists the nominal plant parameters for the accident analyses. The applicant states that uncertainties in initial plant conditions are applied in accordance with the applicable approved methodologies. However, the staff notes that FSAR Tier 2, Section 15.0.0.3.1 does not provide sufficient information of the instrumentation or methodology for the main feedwater flow measurement. The staff also notes that FSAR Tier 2, Section 15.0.0.3.1 does not provide a basis for the statement that the main feedwater flow measurement supports a 0.48 percent power uncertainty. In RAI 432, Question 15.00.02-1, the staff requested additional information from the applicant to provide a description of the mechanism, such as an ITAAC or COL action item, by which the information will be provided to support the 0.48 percent total power measurement uncertainty claimed by the applicant, including a description of how it will be verified and confirmed. **RAI 432, Question 15.00.02-1 is being tracked as an open item**.

15.0.1.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.0.1.6 *Conclusions*

FSAR Tier 2, Chapter 15 follows the format and content of RG 1.206, “Combined License Applications for Nuclear Power Plants, June, 2007 and, therefore, meets 10 CFR Part 52, “Early Site Permits; Standard Design Certification; and Combined Licenses for Nuclear Power Plants.”

The analysis contained in FSAR Tier 2, Section 15.0 meets 10 CFR Part 50, especially 10 CFR 50.46 and Appendix A, as discussed in Section 15.6.5 of this report and the GDC described below.

Based on the discussion above, and with the exception of the open items identified above, the staff finds that the U.S. EPR design meets:

1. GDC 2, as it relates to the seismic design in that equipment relied upon to mitigate an AOO or accident is safety-related and seismically qualified.

2. GDC 4, as it relates to the requirement for environmental qualification of equipment relied upon to mitigate an AOO or accident is safety-related and seismically qualified.

3. GDC 5, as it relates to sharing of equipment amongst units in that safety equipment is not shared.

4. GDC 10, as it relates to ensure that specified acceptable fuel design limits are not exceeded during normal operations including AOOs as embodied in the U.S. EPR AOO and accident acceptance criteria.

5. GDC 15, as it relates to the RCS being designed to ensure that the pressure boundary will not be breached as embodied in the U.S. EPR AOO and accident acceptance criteria.

6. GDC 17, as it relates to the requirement that AOOs and accidents have been analyzed with and without onsite and offsite electric power available.

7. GDC 20, as it relates to the reactivity control systems ensure that the plant does not exceed specified acceptable fuel design limits as embodied in the AOO and accident analysis.

8. GDC 25, as it relates to the reactor protection system ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system, as embodied in the AOO and accident analysis.

9. GDC 26, as it relates control of reactivity changes to ensure that specified acceptable fuel design limits are not exceeded even during AOOs by ensuring appropriate margin for malfunctions such as stuck rods, as embodied in the AOO and accident analysis.

10. GDC 27, and GDC 28, as they relate to the RCS being designed with an appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained, as embodied in the AOO and accident analysis.

11. GDC 29, as it relates to the design of the protection and reactivity control systems and their performance (i.e., to accomplish their intended safety functions) during AOOs, as embodied in the AOO and accident analysis.

12. GDC 34, as it relates to the capability to transfer decay heat and other residual heat from the reactor so that fuel and pressure boundary design limits are not exceeded, as embodied in the AOO and accident analysis.

13. GDC 35, as it relates to the RCS and associated auxiliaries being designed to provide abundant emergency core cooling as embodied in the accident analysis.

The staff finds that the U.S. EPR acceptance criteria shown in FSAR Tier 2, Table 15.0-2, "Accident Analysis Acceptance Criteria," provide the acceptance criteria for AOOs and Accidents include the analysis acceptance criteria shown in the Regulatory Bases section above and provide an acceptable basis by which the U.S.EPR meet the GDC above.

15.0.2 Computer Codes Used in Analysis

Review of this section of the FSAR is documented under Section 15.0.1 of this report.

15.0.3 Radiological Consequences of Design Basis Accidents

15.0.3.1 *Introduction*

The evaluation of those design basis accidents that have radiological consequences are evaluated in this section and compared to the applicable regulatory acceptance criteria.

15.0.3.2 *Summary of Application*

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.6.1, "Main Control Room Air Conditioning System," Section 2.6.3, "Annulus Ventilation System," and Section 5.0, "Site Parameters."

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.0.3, summarized here, in part, as follows:

In U.S. EPR FSAR Tier 2, Chapter 15, the applicant performed radiological consequence assessments of the following seven reactor design basis accidents, using the hypothetical set of atmospheric relative concentration (dispersion) values (χ/Q values) for accidents. Because all other aspects of the design are fixed, these χ/Q values help determine the required minimum distances to the exclusion area boundary (EAB) and the low-population zone (LPZ) for a given site in order to provide reasonable assurance that the radiological consequences of a DBA will be within the siting dose criteria given in regulation, as identified below.

FSAR Tier 2, Section 15.0.3 provides discussion of the DBA radiological consequences analyses. The DBAs analyzed for radiological consequences include:

- Small line carrying primary coolant break outside of the Reactor Building
- Steam generator tube rupture
- Main steam line break outside of the Reactor Building
- Reactor coolant pump locked rotor accident (LRA)
- Rod ejection accident (REA)
- Fuel handling accident (FHA)
- Loss-of-coolant accident

The applicant provided information on the radiological consequences analysis methodology, assumptions, and results for the dose at the EAB, at the LPZ outer boundary, and in the control room. The applicant also provided information on the radiological habitability in the U.S. EPR design technical support center (TSC) to show compliance with the on-site emergency response facility regulatory requirements.

In FSAR Tier 2, Section 15.0.3, the applicant concluded that the U.S. EPR design will provide reasonable assurance that the radiological consequences resulting from any of the above DBAs will fall within the offsite dose criterion of 0.25 Sv (25 rem) total effective dose equivalent (TEDE), as given in 10 CFR 52.47(a)(2), "Contents of applications; technical information," and the control room operator dose criterion of 0.05 Sv (5 rem), as given in 10 CFR Part 50, Appendix A, GDC 19, "Control Room," as incorporated by reference in 10 CFR 52.47(a)(3). The applicant reached this conclusion by performing the DBA radiological consequences analyses by:

- Using reactor accident source terms based on NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," and RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors"
- Crediting control of the pH of the water in the containment to prevent iodine evolution
- Using a set of hypothetical atmospheric dispersion factor (χ/Q) values

The χ/Q values are the relative atmospheric concentrations of radiological releases at the receptor point in terms of the rate of radioactivity release. In lieu of site-specific meteorological data, the applicant provided a reference set of χ/Q values for the U.S. EPR design using

meteorological data that is expected to bound many U.S. operating nuclear power plant sites for offsite dispersion. FSAR Tier 2, Table 2.1-1, "U.S. EPR Site Design Envelope," provides the reference set of χ/Q values for the U.S. EPR design. Accident-specific control room and TSC receptor χ/Q values used in the analyses are also given in FSAR Tier 2, Table 2.1-1.

FSAR Tier 2, Table 15.0-12, "Radiological Consequences of U.S. EPR Design Basis Accidents," summarizes the results from the DBA radiological consequence evaluations and compares these results to the applicable dose acceptance criteria.

ITAAC: There are no ITAAC specific to this area of review.

Technical Specifications: There are no Technical Specifications for this area of review.

15.0.3.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.0.3. The staff also referred to NUREG/CR-5950, "Iodine Evolution and pH Control" in this area of review.

1. 10 CFR 52.47, Subparagraph (a)(2), as it relates to the evaluation and analysis of the offsite radiological consequences of postulated accidents with fission product release.

2. GDC 19, as it relates to maintaining the control room in a safe condition under accident conditions by providing adequate protection against radiation.

3. 10 CFR 100.21, "Non-seismic Siting Criteria," as it relates to the evaluation and analysis of the radiological consequences of postulated accidents for the type of facility to be located at the site in support of evaluating the site atmospheric dispersion characteristics.

4. 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," Paragraph IV.E.8, as it relates to adequate provisions for an onsite technical support center from which effective direction can be given and effective control can be exercised during an emergency.

5. GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," as it relates to isolation of all pipes that are part of the reactor coolant pressure boundary and penetrate the Containment Building.

The related acceptance criteria are as follows:

- Regulatory Guide 1.183

15.0.3.4 *Technical Evaluation*

The staff evaluated the radiological consequences of DBAs against the dose criteria, given in 10 CFR 52.47(a)(2)(iv), of 0.25 Sv (25 rem) TEDE at the EAB for any 2-hour period following the onset of the postulated fission product release, and 0.25 Sv (25 rem) TEDE at the outer

boundary of the LPZ for the duration of exposure to the release cloud. The staff used a criterion of 0.05 Sv (5 rem) TEDE for evaluating the radiological consequences from DBAs in the control room of the U.S. EPR design, pursuant to GDC 19. The staff used guidance in SRP Section 15.0.3 and RG 1.183 in its review of the U.S. EPR DBA radiological consequence analyses. Although RG 1.183 was written to apply to the currently operating power reactors, its guidance on radiological acceptance criteria, formulation of the source term, and DBA modeling is useful in the review of the U.S. EPR design.

The staff evaluated the DBA radiological habitability analysis for the U.S. EPR design TSC against the onsite emergency response facility regulatory requirements in 10 CFR Part 50, Appendix E, Paragraph IV.E.8 and 10 CFR 50.47(b)(8) and (b)(11), "Emergency Plans." The staff's complete review of the emergency response facilities is discussed in Section 13.3, "Emergency Planning," of this report.

The staff reviewed the radiological consequence analyses performed by the applicant using the hypothetical χ/Q values given in FSAR Tier 2, Table 2.1-1. The staff finds that the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria stated above. To evaluate the applicant's analyses, the staff performed independent radiological calculations for the above DBAs using the hypothetical χ/Q values provided by the applicant and the RADTRAD Version 3.03 computer code described in Supplement 2 to NUREG/CR-6604, "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation." The following sections describe the staff's findings.

The applicant followed the accident analysis guidance in RG 1.183 and SRP Section 15.0.3. The U.S. EPR DBA radiological consequences analyses credit safety-related structures, systems, and components to mitigate the radiological consequences of a DBA. If the assumption resulted in a more limiting radiological consequence, non-safety-related SSCs were assumed operational. Any credited operator actions from the main control room were assumed to take place 30 minutes or later after the start of the accident. The analyses evaluated the DBAs considering a single, active failure that maximizes the radiological consequences and additionally assume a loss of offsite power either coincident with the event or with a reactor trip, whichever is limiting. Each analysis assumed the loss of one train of the two redundant charcoal filtrations systems of the main control room for the duration of the accident.

15.0.3.4.1 Accident Source Terms

The U.S. EPR is an evolutionary PWR design. The primary system design and main components design are similar to those of currently operating reactors, and the plant design includes active engineered safety features (ESFs) to mitigate accidents. In SECY-94-302, "Source Term-Related Technical and Licensing Issues Relating to Evolutionary and Passive Light-Water-Reactor Designs," December 19, 1994, the staff proposed to use only the coolant, gap, and early in-vessel releases from NUREG-1465 for the radiological consequence assessments of DBAs for evolutionary and passive light-water reactor designs. These source terms encompass a broad range of accident scenarios, including significant levels of core damage with the core remaining in the vessel. These scenarios define the most severe accidents from which the plant could be expected to return to a safe-shutdown condition. The revised source terms in NUREG-1465 must be applied conservatively in evaluating DBAs in conjunction with conservative assumptions in calculating doses, such as adverse meteorology. Application to severe accidents may use more realistic assumptions.

The staff considered the inclusion of the ex-vessel and the late in-vessel source terms to be unduly conservative for DBA purposes. Such releases would only result from core damage accidents with vessel failure and core-concrete interactions. For evolutionary and passive light-water reactors, the estimated frequencies of such scenarios are low enough that they do not have to be considered credible for the purpose of meeting 10 CFR 50.34, as reiterated in 10 CFR 52.47. In SECY-94-302, the Commission approved the staff-recommended technical positions to use only the coolant, gap, and early in-vessel releases in NUREG-1465 for the radiological consequence assessments of DBAs for evolutionary and passive light-water reactor designs.

The NRC issued RG 1.183 in July 2000, to provide guidance to licensees of operating power reactors on acceptable applications of alternative source terms pursuant to 10 CFR 50.67, "Accident Source Term." This RG provides guidance based on insights from NUREG-1465 and significant attributes of other alternative source terms that the staff may find acceptable for operating light-water reactors (LWRs). It also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted alternative source term for operating power reactors. In SRP Section 15.0.3, the staff's review procedures direct the use of RG 1.183 regulatory positions, as far as applicable to the plant design under review. The applicant followed the relevant guidance in RG 1.183 for PWRs.

For DBAs other than the LOCA and the FHA, the source of radioactive materials available for release is the primary and secondary coolant. The staff's review of the coolant source terms is discussed in Section 11.1, "Source Term," of this report.

The U.S. EPR DBA radiological consequences analyses are based on 100.5 percent of rated core thermal power. The 0.5 percent power uncertainty is based on use of an ultrasonic flow meter to measure the feed water flow rate used in the heat balance calculation of core power. Accordingly, the applicant calculated the core fission product isotopic inventory at 100.5 percent of the core rated thermal power of 4,590 MWt, which is 4,612 MWt. The applicant used the ORIGEN 2.1 isotope generation and depletion computer code, along with extended burnup libraries for ORIGEN 2 high burnup reactor models, to calculate the core isotopic inventory. RG 1.183 states that ORIGEN 2 is an appropriate code for calculation of the core inventory. However, the staff had the following question related to its use:

The staff noted that the ORIGEN 2.1 generation and depletion code was used to calculate the core radionuclide inventory. Oak Ridge National Laboratory (ORNL) does not support the ORIGEN 2 code any longer, but instead recommends use of the ORIGEN-ARP or ORIGEN-S code included in the SCALE code package, which is kept up-to-date. SCALE 5.1 is the latest release and includes libraries for high burnup fuel, up to 72 gigawatt days per metric ton uranium (GWD/MTU). In RAI 17, Question 15.00.03-1, the staff requested that the applicant justify the use of an older unsupported version of the ORIGEN code.

In a July 16, 2008, response to RAI 17, Question 15.00.03-1, the applicant provided an evaluation comparing the ORIGEN 2.1 core inventory against an ORIGEN-S core inventory for the U.S. EPR core. The core inventory was calculated assuming a core enrichment of 5 percent U-235 and a burnup of 62 GWD/MTU. The code comparison was made based on the difference between the LPZ doses calculated for the LOCA containment leakage pathway for each core inventory. The applicant evaluated the sensitivity of the code core inventory calculations for two different sets of principal radionuclides, 84 and 294 nuclides. The applicant's evaluation shows that the ORIGEN 2.1 calculations give higher dose results than a

comparable ORIGEN-S calculation; therefore, the core inventory given in the FSAR calculated using ORIGEN 2.1 is appropriately conservative.

The U.S. EPR core maximum fuel assembly burnup is 62 GWD/MTU. To calculate a bounding core fission product isotopic inventory, the applicant performed a parametric evaluation with fuel enrichment ranging from 2 weight percent (wt%) to 5 wt% U-235 and burnups ranging between approximately 5 and 62 GWD/MTU. Each case assumed continuous reactor operation at full power without any refueling outage. The maximum activity for each radionuclide from any of the cases was chosen to make a composite bounding core fission product inventory for use in the DBA radiological consequences analyses. The core inventory is given in FSAR Tier 2, Table 15.0-14, "Design Basis Core Radionuclide Inventory." The staff requested the following information about the parametric calculations performed by the applicant:

The staff notes that FSAR Section 15.0.3.3.4 states that the fuel burnup ranges between approximately 5 and 62 GWD/MTU. In RAI 17, Question 15.00.03-2, the staff requested that the applicant explain if the cross-section libraries used in the calculation of the core radionuclide inventory were applicable to the maximum fuel burnup assumed.

In RAI 17, Question 15.00.03-3, the staff requested that the applicant provide the length of the operating cycle and the number of cycles that are assumed in the core radionuclide inventory calculations.

In a July 16, 2008, response to RAI 17, Questions 15.00.03-2 and 15.00.03-3, the applicant affirmed that the cross-section libraries for extended burnup PWR fuel, extrapolated to 62 GWD/MTU, were used in the calculation of the core inventory and that the calculation was not dependent on the refueling outage length. The staff performed some limited confirmatory analyses with ORIGEN-ARP using the extended burnup inventories up to 72 GWD/MTU and varying the cycle lengths and was able to determine that the cross-section libraries and assumptions on core operation used by the applicant in the core inventory calculations were acceptable.

Based on the responses to RAI 17, Questions 15.00.03-1, 2, and 3, the staff finds that the applicant has sufficiently addressed the staff's questions on use of the ORIGEN 2.1 code. Based on the above discussion, the staff finds that the applicant's calculation of the core fission product isotopic inventory conforms to RG 1.183 guidance and is acceptable. Therefore, the staff considers RAI 17, Questions 15.00.03-1, 2, and 3 resolved.

15.0.3.4.2 Hypothetical Atmospheric Dispersion Factors

Because no specific site is associated with the U.S. EPR plant, the applicant defined the offsite boundaries only in terms of hypothetical atmospheric relative concentration (χ/Q) values at fixed EAB and LPZ distances. FSAR Tier 2, Table 2.1-1 lists the hypothetical reference accident χ/Q values used in the radiological consequence analyses for the U.S. EPR design. Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases," of this report provides discussion of the staff's review of the hypothetical atmospheric dispersion factors. The staff will perform an independent assessment of short-term (less than or equal to 30 days) atmospheric dispersion factors for potential accident consequence analyses on a site-specific basis for a COL application that references the U.S. EPR design.

To determine the most conservative dose for each receptor, the applicant used time-shifting of the reference accident χ/Q values when input to the dose calculation. The input atmospheric

dispersion factor values were shifted in time so that the most adverse release of radioactive material to the environment occurs coincident with the period of most adverse atmospheric dispersion. The largest reference accident χ/Q value for each release/receptor pair is that for the 0-2 hour period. In conformance with the guidance in RG 1.183, Regulatory Position 4.1.5, and RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Regulatory Position 2, if the period of the highest activity release occurs later than 2 hours, the 0-2 hour χ/Q value would be input at a later time to coincide with the largest release. For example, the applicant's LOCA analysis shifted the 0-2 hour reference accident χ/Q values for the EAB, LPZ, and control room receptors to be applied for the time period 1.5 hours to 3.5 hours. For the LPZ and control room receptors, the 2-8 hour value was used for the time periods before and after the peak release (i.e., 0 to 1.5 hours and 3.5 to 8 hours). The rest of the reference accident χ/Q values were used for their nominal time periods.

15.0.3.4.3 Post-Accident Containment Water Chemistry Management

Management of the post-accident containment water chemistry must comply with the requirements of GDC 41, "Containment Atmosphere Cleanup," and GDC 4, "Environmental and Dynamic Effect Design Bases." By minimizing the release of radioactive iodine from the containment sump water, the water chemistry will meet the requirement of GDC 41, as it relates to the ability of the design of containment atmosphere cleanup systems to control fission product releases to the reactor containment following postulated accidents. By preventing stress-corrosion cracking of stainless steel components exposed to the water accumulated in the containment sump, the water chemistry will meet the requirement of GDC 4 that components important to safety be compatible with the environmental conditions associated with accident conditions, including LOCAs.

NUREG-1465 states that after an accident, iodine entering the containment from the reactor core is composed of at least 95 percent cesium iodide (CsI), with the remaining five percent elemental iodine (I_2) and a small amount of hydrochloric acid. However, about three percent of elemental iodine in contact with some organic compounds will produce organic iodides. Therefore, the iodine in the containment will consist of 95 percent particulate iodine as CsI, 4.85 percent elemental iodine, and 0.15 percent organic iodine. The composition of the iodine in the U.S. EPR conforms to the composition stated in NUREG-1465.

Iodine in the form of CsI is soluble in the containment water. However, some of it may be converted into the elemental form, which is considerably less soluble, and will be released into the containment atmosphere.

The released radioactive iodine may leak out of the containment and contribute to outside radiation doses. To minimize formation of the elemental iodine, the pH of the containment water should be kept basic. Basic pH will also prevent stress-corrosion cracking of the stainless steel components.

Verification of Post LOCA Sump pH

Background

RG 1.183, Appendix A, "Assumptions for Evaluating the Radiological Consequences of a LWR Loss-of-Coolant Accident," Item 2 identifies the need for the containment sump pH to be greater than 7.0 if the assumptions for iodine partitioning in the regulatory guide are acceptable for use.

Design requirements for the plant undergoing design certification, including the mitigating features such as the containment, are found in 10 CFR 52.47 which identifies that the fission product release for a major accident is the source term for determining adequacy of design.

FSAR Tier 2, Section 15.0.3.12, "Postaccident Reactor Building Water Chemistry Control," describes the water chemistry of the containment sump following the design basis accident. The post-LOCA functions of the containment water chemistry are also described in FSAR Tier 2, Sections 6.5.3.1, "Primary Containment," and 6.5.3.2, "Secondary Containment."

FSAR Tier 2 Section 15.0.3.2, "Event Categorization," identifies the LOCA as one of the seven DBAs that will result in a radiological consequence outside the confines of the plant Containment Building. The mass of trisodium phosphate (TSP) dodecahydrate identified in FSAR Tier 2, Section 15.0.3.12.3, "Results," is 5,534 kg (12,200 lb) to maintain pH 7.0 or greater. FSAR Tier 2, Tables 15.0-55, "H^+ Added to IRWST," and 15.0-56, "Mass of TSP vs. pH at 30 Days," provide the results of the calculations for demonstrating that a pH 7.0 is maintained for 30 days considering the formation of nitric and hydrochloric acids in containment resulting from degradation of cable jacketing materials. The applicant did not provide the calculational methodology used to demonstrate that the pH of the recirculating water from the in-containment refueling water storage tank (IRWST) to the core remains above 7.0 for the duration of the accident. Therefore, in RAI 137, Question 15.00.03-33, the staff requested that the applicant provide the estimated containment sump pH for the 30-day period following a LOCA and the data used to make these estimates. In a January 28, 2009, response to RAI 137, Question 15.00.03-33, the applicant provided the requested information. However, the staff performed an independent calculation to verify that the pH of the recirculating water will remain above 7.0 for the duration of the accident. In follow-up RAI 206, Question 15.00.03-36, the staff requested that the applicant provide containment and containment sump activity/dose information that could affect the concentrations of strong acids (e.g., HC, HNO_3) in the containment sump that affect pH. In an April 23, 2009, response to RAI 206, Question 15.00.03-36, the applicant provided details regarding the G-value used for the nitric acid formation and its calculations on hydrochloric acid production during the post-LOCA event. The values provided have been independently verified, and results of the final calculation were found to be reasonable based on the information provided. In its supplemental response to RAI 137, Question 15.00.03-33, the applicant provided Tables 15.00.03-33-1 and 15.00.03-33-3 to demonstrate that the pH would be acceptable after the 30-day time frame with the added amounts of HCl and HNO_3 resulting from radiolytic reactions. These tables were not included in the FSAR. Therefore, in follow-up RAI 394, Question 15.00.03-37, the staff requested that the applicant include these tables in FSAR Tier 2, Section 15.0.3. **RAI 394, Question 15.00.03-37 is being tracked as an open item**.

Methodology

The staff used Electric Power Research Institute (EPRI) software (Multi-Q) to perform the independent calculation of the post-LOCA pH. In order to ensure that the post-LOCA pool pH will be greater than 7.0, the TSP mass required must be calculated based on the maximum boron concentration that can be present in all systems and components contributing to the containment pool at the minimum volume. Table 15.0.3-1, "Input Parameters for the Containment Sump at 1 Day after the LOCA," below identifies these parameters based on elements identified in the FSAR. The sodium concentration is based on the stoichiometric amount of sodium in trisodium phosphate dodecahydrate, and the lithium concentration is based on the lowest lithium allowed by the EPRI program at the estimated beginning of cycle reactor

coolant system boron concentration (1,400 ppm) just prior to the LOCA. The determination of the quantity of strong acid produced is based on the G-values for formation of nitric acid and hydrochloric acid in water saturated with nitrogen and cable insulation exposed to direct radiation from the containment atmosphere. The staff used the NUREG/CR-5950, "Iodine Evolution and pH Control," methodology to perform the calculation of the generation of strong acids. The G-value is the amount of energy per second required to form a molecule of the species under investigation. Knowing the total decay energy produced in the water, air phase of the containment, and time period of exposure, the buildup of the respective species may be calculated.

Table 15.0.3-1 Input Parameters for the Containment Sump at 1 Day after the LOCA

	IRWST	RCS	PZR	Accumulators (4)	Extra-Borating Tanks (2)	Final Containment
Reference	FSAR Tier 2	FSAR Tier 2		FSAR Tier 2	FSAR Tier 2	Recirc Water
	Chap 16, TS 3.5.4	Chapter 5 Table 5.1-1		Chap 16, TS 3.5.1	Chap 16, TS 3.5.5	
Volume, m^3	1,894	385	75	140	67	
Volume, ft^3	66,886	13,596	2,649	4,940	2,345	
Volume, L	1,892,500	384,927	74,998	139,860	66,391	2,558,676
Volume, gal	500,000	101,698	19,815	36,951	17,541	676,005
Boron concentration ppm, max	1,900	1,400	1,400	1,900	7,300	1,950
TSP mass, kg						5,534
TSP mass, lb						12,200
TSP concentration, M						0.013
Phosphate concentration, ppm						1,269
Cl^- from Hypalon, ppm (at 1 d)						32
Cl^- from PVC, ppm (at 1 d)						4.6
NO_3^-, ppm (at 1d)						1.16
Na, ppm (at 1 d)						911
Li, ppm						0.54
Si, ppm as H4Si208						0.1
F^-, ppm						0.005
SO_4, ppm						0.005
NH_3, ppm						0.04

Results

The values for silica, sulfate, ammonia, and fluoride were not provided in FSAR Tier 2, Table 15.0-54, "IRWST pH Analysis Inputs"; therefore, the staff requested that the applicant provide the values used in the calculation in RAI 137, Question 15.00.03-34. In a January 28, 2009, response to RAI 137, Question 15.00.03-34, the applicant indicated that the total chloride and nitrate concentrations would be the same as those calculated from the degradation of hypalon cable coverings and nitrogen radiolysis in water. Concentrations of other contaminants were constant throughout the 30-day period because:

- The only contribution to sodium and phosphate is from the TSP additive.
- No calcium silicate based insulation will be used inside containment (thus these contaminant levels will be very low and not contribute to the total ionic content of the sump water).

The phosphate concentration credited in the RAI 137, Question 15.00.03-34 response by the applicant is 0.00607 moles PO_4/liter. Independent calculations using Table 15.0.3-1 parameters (above) find the minimum value of the phosphate concentration to be 0.00547 moles PO_4/liter based on 5,534 kg (12,200 lb) of TSP, maximum boron concentration of 1,948 ppm (equivalent to boric acid concentration of 11,407 ppm), and a maximum volume of water of 2,668,754 L (705,000 gal). When the above Table 15.0.3-1 and impurity parameters are input into Multi-Q, a final pH at the end of the first day is 7.15 standard units at 25 °C (77 °F), when the maximum volume of all contributing systems is used. This value represents the pH minimum for the post-LOCA sump, because the maximum boron concentration from each system is used with the maximum volume. The molar quantities of acid produced over a 30-day period do not affect the calculation of this pH value due to the buffering capacity of the phosphate. This means that the pH will not change over 30 days based on the additional HCl or HNO_3 that is formed due to radiolysis.

This pH value is lower than the value of 7.5 identified in the FSAR Tier 2, Table 15.0-56. Part of the difference is due to the bounding volumes and concentrations used in this independent calculation. The pH from this independent calculation is 7.15 standard units. Although this is lower than the value of 7.5 standard units cited by the applicant, it puts a lower bound on the pH that is still within an acceptable range.

On the basis of the results discussed above, the staff determined that the applicant's response to RAI 137, Question 15.00.03-34 is acceptable. Therefore the staff considers RAI 137, Question 15.00.03-34 resolved.

Technical Specifications

FSAR Chapter 16, Technical Specifications 3.6.3 "Containment Isolation Valves" (isolation to prevent release of radioactivity) and 3.6.8 "pH Adjustment" (pH adjustment chemical TSP volume) both relate to the conditions of FSAR Tier 2, Section 15.0.3. There are no chemical requirements in TS 3.6.3.

The limiting condition of operation (LCO) for TS 3.6.8 is that the pH adjustment baskets shall contain ≥ 5.97 m^3 (211 ft^3) of trisodium phosphate, and the frequency of determination is to be once per 24 months. The FSAR Tier 2, Section 15.0.3.12.3 identifies the mass of TSP as 5,534 kg (12,200 lb). Converting the TS 3.6.8 volume into a mass based on the standard handbook value of density of TSP (the dodecahydrate salt is the compound identified in the FSAR) of 1.6 g/mL (100 lb_m/ft^3) yields 9,545 kg (21,043 lb). This number is in conflict with the FSAR stated mass of 5,534 kg (12,200 lb) unless certain assumptions are made regarding the density and purity of the TSP. Therefore, in RAI 137, Question 15.00.03-35, the staff requested additional information. The independent calculation identified above shows that the mass of 5,534 kg (12,200 lb) of trisodium phosphate dodecahydrate is sufficient to maintain the post-LOCA pH above 7.1. In a January 28, 2009, response to RAI 137, Question 15.00.03-35, the applicant indicated that the required mass of 5,534 kg (12,200 lb) was calculated based on a manufactured density of the TSP of 0.93 g/mL (58 lb_m/ft^3). This provides a conservative value for the mass of TSP, since the volume in containment is the measurement made to fulfill the

surveillance requirement (TS B3.6.8) of a minimum volume of 5.975 m^3 (211 ft^3). [Note: Mass = volume/density. A lower density at a given required volume provides a higher mass value ensuring pH will be in the range required by the TS surveillance.] Additionally, the applicant stated that the purchase specifications will require the purity and density of the TSP to be stated on each shipment so that the assurance of minimum neutralizing capability will be maintained. The staff finds the applicant's response to RAI 137, Question 15.00.03-35 to be acceptable. However, the staff determined that the purity and density values for commercially purchased TSP should be included by the applicant in FSAR Tier 2, Section 15.0.03. The staff issued a follow-up question to request that the necessary information be added to FSAR Tier 2, Section 15.0.03. **RAI 394, Question 15.00.03-38, which is associated with the above request, is being tracked as an open item**.

Summary

The applicant has identified the design features and general design criteria for control of radioiodines in the containment following a LOCA. The FSAR Tier 2, Sections 15.0.3.12 and 6.5.2, "Containment Spray Systems," provide sufficient detail to be in conformance with the guidance in NUREG-0800, Section 15.0.3.

Normal and abnormal conditions of an environmental and plant related nature have been accounted for in the design of the containment IRWST and ventilation systems.

Trisodium phosphate dodecahydrate is identified as the chemical used to maintain IRWST pH at greater than 7.0 standard units to minimize airborne radioiodines and thus minimize doses to the public from releases and to control room operators during the accident.

Subject to the resolution of RAI 89, Questions 15.00.03-37 and 15.00.03-38, which are being tracked as open items, the staff concludes that the information contained in the FSAR conforms to the guidance of RG 1.183 regarding control of the radioiodines in the post LOCA environment as related to the source term iodine assumptions for use in DBA radiological consequences analyses. Therefore, the staff finds the information contained in FSAR Tier 2, Section 15.0.3.12 acceptable.

15.0.3.4.4 Radiological Consequences of Small Line Carrying Primary Coolant Break Outside of the Reactor Building

GDC 55 contains a provision to ensure isolation of all pipes that are part of the reactor coolant pressure boundary and penetrate the Containment Building. GDC 55 also provides that small-diameter pipes that must be continuously connected to the primary coolant system to perform necessary functions may be acceptable based on some other defined bases. For these lines, methods of mitigating the consequences of a rupture are necessary, because the lines cannot be isolated automatically. For the U.S. EPR design, the applicant determined that the bounding small lines in this category are one of three nuclear sampling system (NSS) 1/4 inch sampling lines to the RCS and the chemical volume control system 6-inch line between the volume control tank (VCT) and VCT suction valves.

A break in these lines requires manual action for isolation, while other small breaks are automatically isolated or release a smaller amount of primary coolant. No instrument lines carry primary coolant outside containment in the U.S. EPR design. The applicant submitted a radiological analysis for a small line failure in FSAR Tier 2, Section 15.0.3.5, "Small Line Carrying Primary Coolant Break Outside of the Reactor Building Accident." Additional

description of the small line break outside of containment was given in FSAR Tier 2, Section 15.6.2, “Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment.”

The flow from either line is passively restricted by the size of the line itself, and given the same isolation time by manual action at 30 minutes, the CVCS line break would lead to a much larger coolant mass release than the NSS line break. The CVCS line break is downstream of the purification system, which filters iodine and reduces the temperature to 51.7 °C (122 °F), which leads to no coolant flashing at the CVCS break. Therefore, iodine release to the environment is relatively insignificant. However, because of the larger coolant mass release, the CVCS break results in a significantly larger release of noble gases than for the NSS line break. Although the NSS line break has a smaller coolant mass release, the temperature of the coolant as it is released is greater than 293.3 °C (560 °F), which would cause flashing of the coolant to steam and increased release of iodine as compared to the CVCS line break. The applicant determined that the NSS line break analysis dose results at the EAB, LPZ, control room, and TSC are bounding for the design basis small line breaks.

For both cases, fuel damage is not assumed, because the loss of primary coolant is relatively small and is compensated by the safety injection system. The primary coolant activity concentrations are assumed to be initially at the proposed TS 3.4.15, “RCS Specific Activity,” RCS maximum equilibrium limits for continued operation. The applicant assumed an accident-initiated iodine spike in the primary coolant is caused by the postulated reactor shutdown or depressurization. The iodine spike raises the equilibrium iodine appearance rate by a factor of 500, in accordance with the guidance in RG 1.183. The applicant’s analysis considers the release of halogens (including iodine), noble gases, and alkali metals. Other radionuclides in the released coolant are assumed to remain in the liquid phase.

The fraction of the halogens and alkali metals in the released coolant that becomes airborne and available for release to the atmosphere is assumed to be equal to the fraction of the coolant that flashes to steam, as determined by assuming a constant enthalpy process. Noble gases released from the RCS are assumed to be released directly to the environment without mitigation. The staff finds the assumptions on the amount of released coolant fission products that are assumed to be airborne acceptable and in agreement with guidance on assumptions for radioactivity released from a small line break found in SRP Section 15.6.2, “Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment.”

To verify the applicant’s assessment, the staff performed independent radiological consequence calculations for a postulated small line break accident using the applicant’s assumptions on accident progression, fission product transport, and design reference atmospheric dispersion factors, and guidance in SRP Section 15.6.2 and RG 1.183 as applicable to the small line break accident. The offsite radiological consequences calculated by the staff conform to those calculated by the applicant and confirm that the small line break doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the small line break.

Based on comparison of the applicant’s analysis methodology to guidance in SRP Section 6.5.2 and RG 1.183, and on the results of the staff’s confirmatory dose analysis, the staff finds that the applicant’s analyses of the radiological consequences of the design basis small line break are acceptable. The staff concludes that the U.S. EPR design, as bounded by the atmospheric relative concentrations proposed by the applicant, provides reasonable assurance that the

radiological consequences of a postulated small line break will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 52.47.

FSAR Tier 2, Section 15.0.3 reports the results of the applicant's radiological consequence analysis for personnel in the main control room and TSC during a design basis small line break accident, which relies on the control room air conditioning system (CRACS) to limit the radioactivity to which the personnel may be exposed. Section 6.4, "Habitability Systems," of this report describes the staff's review and assessment of control room habitability, and Section 9.4, "Air Conditioning, Heating, Cooling, and Ventilation Systems," of this report describes the staff's review and assessment of the CRACS design. To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the small line break accident, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The control room and TSC radiological consequences calculated by the staff conform to those calculated by the applicant and confirm that the small line break doses calculated by the applicant are less than 0.05 Sv (5 rem) TEDE. Therefore, the staff has determined that there is reasonable assurance that the CRACS can mitigate the dose in the main control room and TSC following a design basis small line break accident to meet the dose criterion given in GDC 19.

15.0.3.4.5 Radiological Consequences of Steam Generator Tube Rupture

The applicant has evaluated the radiological consequences of a postulated SGTR accident and provided a radiological consequence analysis for the accident in FSAR Tier 2, Section 15.0.3.6, "Steam Generator Tube Rupture Accident." Additional description of the SGTR was given in FSAR Tier 2, Section 15.6.3, "Steam Generator Tube Failure (PWR)." This DBA assumes that a single tube in one steam generator fails, releasing primary coolant to the secondary side of the affected SG. The analysis assumes a coincident LOOP and the MSRCV of the affected SG fails in the fully open position. The applicant evaluated four potentially limiting SGTR thermal-hydraulic responses to identify the bounding scenario. Adequate core cooling precludes fuel failure. Following the guidance in RG 1.183, the applicant considered two coolant activity concentration cases.

For Case 1, the analysis assumed that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by the SGTR. Before the postulated accident, the U.S. EPR reactor was assumed to operate at the TS 3.4.15 equilibrium iodine concentration limit of 37 kBq/gm (1.0 μCi/gm) for dose equivalent iodine (DEI)-131 in the primary coolant. The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 335, resulting in a rising iodine concentration in the primary coolant during the course of the accident. For Case 2, the analysis assumed that previous reactor operation had resulted in a primary coolant concentration equal to the maximum instantaneous concentration limit of 2.2 MBq/gm (60 μCi/gm) for DEI-131, given in TS 3.4.15. In each case, the applicant's analysis assumed that halogens, noble gases, and alkali metals contained in the primary coolant released through the ruptured steam generator's broken tube are released directly to the environment until the MSRIV on the ruptured generator is closed at 42.8 minutes after onset of the accident. The three unaffected SGs provide steaming until cold shutdown is maintained. The applicant's analysis conservatively assumed that steaming continued for 30 days. The analysis assumed 0.473 L/min (0.125 gpm) of primary-to-secondary leakage through any SG, which bounds the maximum allowable SG primary-to-secondary leakage as given in the U.S. EPR TS. This

SG leakage carries the activity released to the primary coolant into the secondary coolant. The halogen and alkali fission products are released to the environment through the steaming of the secondary coolant. The steam release is through the main condenser and stack for the first 30 minutes, and through the main steam relief trains and silencers after that time until the end of the plant cooldown.

The staff has reviewed the applicant's analysis and finds that the methods used for the radiological consequence assessment are in conformance with the guidance in RG 1.183, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria.

To verify the applicant's assessments, the staff performed independent radiological consequence calculations for the two coolant activity scenarios for the SGTR accident, following the guidance in RG 1.183 and the applicant's assumptions on accident progression, fission product source terms and transport, and design reference atmospheric dispersion factors. The offsite radiological consequences calculated by the staff conform to those calculated by the applicant for both coolant activity cases, and confirm that the doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the SGTR. Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the applicant's analysis of the design basis SGTR is acceptable. The staff concludes that the U.S. EPR design, as bounded by the atmospheric relative concentrations proposed by the applicant, provides reasonable assurance that the radiological consequences of a postulated SGTR accident with accident-induced iodine spiking will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 52.47.

The staff concludes that the U.S. EPR design, as bounded by the atmospheric relative concentrations proposed by the applicant, provides reasonable assurance that the radiological consequences of a postulated SGTR accident with the reactor coolant at the TS maximum value of 2.2 MBq/gm (60 µCi/gm) for DEI-131 will not exceed the dose criterion set forth in 10 CFR 52.47 (i.e., 0.25 Sv (25 rem) TEDE).

The applicant reported the results of its radiological consequence analysis for personnel in the main control room and TSC during a design basis SGTR, which relies on the CRACS to limit the radioactivity to which personnel may be exposed. To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the SGTR accident, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The control room and TSC radiological consequences calculated by the staff conform to those calculated by the applicant and confirm that the SGTR doses calculated by the applicant are less than 0.05 Sv (5 rem) TEDE for both coolant activity concentration cases.

Therefore, the staff has determined that there is reasonable assurance that the CRACS can mitigate the dose in the main control room and TSC following a SGTR accident to meet the dose criterion given in GDC 19.

15.0.3.4.6 Radiological Consequences of Main Steam Line Break Outside of the Reactor Building

The applicant has evaluated the radiological consequences of a postulated MSLB accident occurring outside of the containment and upstream of the main steam isolation valves. The applicant submitted a radiological analysis for the MSLB accident in FSAR Tier 2, Section 15.0.3.7, "Main Steam Line Break Outside of Reactor Building Accident." Additional description of the MSLB was given in FSAR Tier 2, Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)." The radiological consequences of the design basis MSLB are bounding for feedwater line breaks, as discussed in FSAR Tier 2, Section 15.2.8, "Feedwater Line Breaks Inside and Outside Containment." The limiting MSLB is a double-ended guillotine break of a main steam line, upstream of the MSIV, in the valve compartment in Safeguard Building Division 4. Because the break is upstream of the MSIV, a failure of that MSIV to close was not considered as a single failure. The MSLB does not assume coincident LOOP, because the scenario with available offsite power leads to more induced uncovering of the tubes. The releases to the atmosphere occur through the main steam relief trains and the silencers of the unaffected steam generators, and through Canopy Point 1 for the SG with the broken steam line. The releases through the unaffected SGs end in 8 hours when the residual heat removal system cuts in, and the releases from the affected SG end in 9 hours when the RCS temperature drops below 100 °C (212 °F).

The applicant analyzed this hypothetical accident for four source term cases:

- A pre-accident iodine spike in the primary coolant
- An accident-initiated concurrent iodine spike in the primary coolant
- DNB- induced clad failure
- FCM-induced fuel melting

The applicant performed the clad failure and fuel melting cases to determine the maximum amount of fuel cladding failure or fuel melting for an MSLB, independent of each other, that can be accommodated and not exceed 90 percent of the dose acceptance criterion at any receptor. The applicant's calculations resulted in the assumption of 3.3 percent of the core experiencing fuel rod clad failure in the clad failure case, and the assumption of 0.58 percent of the core melting for the fuel melt case. The staff has reviewed the applicant's analysis and finds that the methods used for the radiological consequence assessment conform to RG 1.183 guidance, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant's assessment, the staff performed independent radiological consequence calculations for the four scenarios for the MSLB accident. The staff's analyses followed guidance in RG 1.183 and used the applicant's assumptions on accident progression, fission product source terms and transport, and design reference atmospheric dispersion factors. The offsite TSC radiological consequences calculated by the staff conform to those calculated by the applicant for all coolant activity and fuel damage cases, and confirm that the doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the MSLB.

Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the applicant's analysis of the design basis MSLB is acceptable. The staff concludes that the U.S. EPR design, as bounded by the

atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated MSLB accident with accident-induced iodine spiking will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 50.47.

The staff also concludes that the U.S. EPR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated MSLB accident for any of the cases with 1) the reactor coolant at the TS maximum value of 2.2 MBq/gm (60 μCi/gm) for DEI-131, 2) 3.3 percent fuel rod clad failure, or (3) 0.58 percent fuel melting, will not exceed the dose criterion set forth in 10 CFR 50.47 (i.e., 0.25 Sv (25 rem) TEDE).

The applicant reported the results of its radiological consequence analysis for personnel in the main control room and TSC during a design basis MSLB, which relies on the CRACS to limit the radioactivity to which the personnel may be exposed. To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the MSLB, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The control room and TSC radiological consequences calculated by the staff conform to those calculated by the applicant and confirm that the MSLB doses calculated by the applicant are less than 0.05 Sv (5 rem) TEDE for each of the source term cases. Therefore, the staff has determined that there is reasonable assurance that the CRACS can mitigate the dose in the main control room and TSC following a design basis MSLB to meet the dose criterion given in GDC 19.

15.0.3.4.7 Radiological Consequences of Locked Rotor Accident

The applicant submitted a radiological analysis for the LRA in FSAR Tier 2, Section 15.0.3.8, "Locked Rotor Accident." The design basis LRA analysis is bounding for the postulated reactor coolant pump shaft break. Additional description of the reactor coolant pump shaft seizure and shaft break was given in FSAR Tier 2, Sections 15.3.3, "Reactor Coolant Pump Rotor Seizure," and 15.3.4, "Reactor Coolant Pump Shaft Break." The locked rotor accident assumes instantaneous seizure of an RCS pump rotor, which leads to a reactor trip on the RCS low flow signal, coincident with LOOP. The MSRTs open to release secondary side steam. The MSRTs release the activity to the environment until the time at which plant cooldown is switched from SG steaming to the RHR. One of the MSRCVs is assumed to fail in the open position. Steam releases continue from the MSRT with the failed open MSRCV until the MSRIV closes. For 15 minutes after the RCS pump rotor seizes, the tubes are uncovered in the SG connected to the main steam line with the stuck open MSRCV, and the primary-to-secondary leakage flashes to steam and is released to the environment without iodine partitioning.

The applicant analyzed this hypothetical accident assuming that 9.5 percent of the core's fuel elements will experience DNB-induced cladding failure, thereby releasing the entire fission product inventory in the fuel-cladding gap of these elements to the reactor coolant. The cladding failure assumption is based on the applicant's determination of the maximum amount of fuel cladding failure that can be accommodated for this accident without exceeding 90 percent of the dose acceptance criteria at any receptor. The primary coolant concentration is assumed to be at the U.S. EPR proposed TS maximum value of 2.2 MBq/gm (60 μCi/gm) for DEI-131 (i.e., pre-accident iodine spike). The analysis assumed 0.473 L/min (0.125 gpm) of primary-to-secondary leakage through any SG, which bounds the maximum allowable SG

primary-to-secondary leakage as given in the U.S. EPR TS. This SG leakage carries the activity released to the primary coolant into the secondary coolant. Steaming of the secondary coolant through three unaffected SGs releases the halogen and alkali fission products to the environment until the end of plant cooldown at 8 hours. The fourth SG releases the entire secondary coolant activity concentration and the primary-to-secondary leakage transported halogens and alkali metals directly to the environment for the first 15 minutes during the time that the affected SG tubes are uncovered.

The staff has reviewed the applicant's analysis and finds that the methods used for the radiological consequence assessment conform to RG 1.183 guidance, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant's assessment, the staff performed independent radiological consequence calculations for the four scenarios for the locked rotor accident. The staff's analyses followed guidance in RG 1.183 and used the applicant's assumptions on accident progression, fission product source terms and transport, and design reference atmospheric dispersion factors. The offsite radiological consequences calculated by the staff conform to those calculated by the applicant for all coolant activity and fuel damage cases, and confirm that the doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the LRA.

Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the applicant's analysis of the design basis LRA is acceptable. The staff concludes that the U.S. EPR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated reactor primary coolant pump seizure accident will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 50.47.

The applicant reported the results of its radiological consequence analysis for personnel in the main control room and TSC during a design basis locked rotor accident, which relies on the CRACS to limit the radioactivity to which the personnel may be exposed. To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the LRA, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The control room and TSC radiological consequences calculated by the staff conform to those calculated by the applicant and confirm that the LRA doses calculated by the applicant are less than 0.05 Sv (5 rem) TEDE. Therefore, the staff has determined that there is reasonable assurance that the CRACS can mitigate the dose in the main control room and TSC following a design basis LRA to meet the dose criterion given in GDC 19.

15.0.3.4.8 Radiological Consequences of Rod Ejection Accident

The applicant submitted a radiological consequence analysis for the REA in FSAR Tier 2, Section 15.0.3.9, "Rod Ejection Accident." Additional description of the REA was given in FSAR Tier 2, Section 15.4.8, "Spectrum of Rod Ejection Accidents in a PWR." The radiological consequences of the design basis REA are bounding for the set of reactivity insertion accidents discussed in FSAR Tier 2, Section 15.4, "Reactivity and Power Distribution Anomalies." The mechanical failure of a control rod mechanism pressure housing is postulated to result in the ejection of a rod cluster control assembly and drive shaft. Because of the resultant opening in the pressure vessel, primary coolant is lost to the containment with concurrent rapid

depressurization of the reactor pressure vessel. This mechanical failure causes a rapid positive reactivity insertion, together with an adverse core power distribution, possibly leading to localized fuel rod damage. The analysis assumes a LOOP coincident with the REA.

In accordance with the guidance in RG 1.183, the applicant evaluated the following two release pathways independently:

- Primary containment leakage pathway, which assumes that that the entire activity released from the fuel becomes airborne in the primary containment and available for release through containment leakage

- Secondary side leakage pathway, which assumes that the entire activity released from the fuel is retained in the RCS and is available for release through SG tube leakage from the primary coolant to the secondary coolant and secondary side steaming

The applicant has assumed two different fuel cladding damage and fuel overheat cases for each of the release pathway scenarios, for a total of four cases. The cladding failure and fuel overheat assumptions are based on the applicant's determination of the maximum amount of fuel cladding failure and fuel overheating (with transient release of fission products from the fuel pellets) as a percentage of the total core that can be accommodated for this accident without exceeding 90 percent of the dose acceptance criteria at any receptor. The four cases are:

- Containment leakage with 33.4 percent fuel clad failure and no fuel overheat

- Containment leakage with 28.6 percent fuel clad failure and 1.14 percent fuel overheat

- Secondary side leakage with 36.7 percent fuel clad failure and no fuel overheat

- Secondary side leakage with 27.8 percent fuel clad failure and 1.11 percent fuel overheat

Although the applicant assumed some fuel overheat in two of the cases, the applicant states that the U.S. EPR design ensures that the post-REA peak fuel temperature will remain below incipient fuel-melting conditions, in conformance with the guidance in SRP Section 4.2, Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents." The staff's review of the fuel system design is discussed in Section 4.2, "Fuel System Design," of this report.

The radioactivity release from fuel cladding failure assumed that the fuel rods have been operating at a radial peaking factor of 1.7, with 10 percent of the fuel rod noble gas and halogen inventory and 12 percent of the alkali metal inventory in the gap initially, which conforms to the guidance in RG 1.183. The release from the fuel for both the fuel cladding failure and the fuel overheat is assumed to occur instantaneously at the initiation of the accident. The release from the overheated fuel consists of the total fuel rod fission product inventory. This assumption exceeds the guidance in RG 1.183 for the amount of fission products initially in the gap for fuel experiencing cladding failure, but is consistent with assumptions on melted fuel. In RAI 17, Question 15.00.03-17, the staff requested that the applicant discuss and justify whether the methodology for calculating the clad failure and fuel overheat accounts for the transient fission product release, as discussed in SRP Section 4.2, Appendix B. In a July 16, 2008, response to RAI 17, Question 15.00.03-17, the applicant stated that this assumption on the release from the overheated fuel is intended to meet the SRP Section 4.2, Appendix B criteria on release from

reactivity insertion accident transient conditions. The staff finds the applicant's assumptions on release from overheated fuel bounds the guidance in SRP Section 4.2, Appendix B, and is therefore acceptable.

The applicant assumed that the release of fission products to the environment may occur via either of two pathways. The containment leakage first pathway involves a release of primary coolant to the containment, which is assumed to leak into the environment at the design leak rate of the containment. The applicant's analysis took credit for aerosol natural deposition and elemental iodine removal in the containment, using the same assumptions as in the LOCA analysis. The staff's review of the containment aerosol deposition and iodine removal is discussed below in Section 15.0.3.10 of this report which documents the staff's review of the applicant's radiological analysis.

In the secondary side leakage pathway, fission products would reach the secondary coolant via the SGs with 0.473 L/min (0.125 gpm) of primary-to-secondary leakage through any SG, which bounds the maximum total allowable primary-to-secondary leakage as given in the U.S. EPR TS. For both pathways, the applicant assumed that the U.S. EPR reactor operated at its TS maximum primary coolant concentration value of 2.2 MBq/gm (60 µCi/gm) for DEI-131.

The staff has reviewed the applicant's analysis and finds that the methods used for the radiological consequence assessment conform to RG 1.183, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant's assessment, the staff performed independent radiological consequence calculations for the four scenarios for the rod ejection accident. The staff's analyses followed guidance in RG 1.183 and used the applicant's assumptions on accident progression, fission product source terms, and design reference atmospheric dispersion factors. The offsite, control room and TSC radiological consequences calculated by the staff conform to those calculated by the applicant for all fuel damage cases, and confirm that the doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the REA.

Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the applicant's analysis of the design basis REA is acceptable. The staff concludes that the U.S. EPR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated rod ejection accident will fall well within the dose criterion set forth in 10 CFR 50.47 (i.e., 25 percent or 0.063 Sv (6.3 rem) TEDE).

The applicant reported the results of its radiological consequence analysis for personnel in the main control room and TSC during a design basis rod ejection accident, which relies on the CRACS to limit the radioactivity to which the personnel may be exposed. To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the REA, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The control room and TSC radiological consequences calculated by the staff conform to those calculated by the applicant and confirm that the REA doses calculated by the applicant are less than 0.05 Sv (5 rem) TEDE for each of the source term cases. Therefore, the staff has determined that there is reasonable assurance that the CRACS can mitigate the dose in the main control room and TSC following a design basis REA to meet the dose criterion given in GDC 19.

15.0.3.4.9 Radiological Consequences of Fuel Handling Accident and Cask Drop

In FSAR Tier 2, Section 15.0.3.10, "Fuel Handling Accident," the applicant presented its analysis of the radiological consequences of a postulated FHA. For the U.S. EPR design, an FHA can be postulated to occur either inside the open containment or in the Fuel Building. The analysis assumptions are bounding for the FHA in either location. The U.S. EPR TS do not require containment or fuel building closure during movement of fuel that has decayed more than 34 hours, and fuel movement is administratively prevented until 34 hours after shutdown. In accordance with guidance in RG 1.183, the applicant assumed that fission products are released directly to the environment within a 2-hour period without credit for any iodine removal processes.

Other fuel handling accidents such as a spent fuel cask falling or tipping onto the spent fuel pool are prevented by the design of the spent fuel handling equipment. The spent fuel cask and transfer machine are located so that a cask is prevented from being in the spent fuel pool area altogether. FSAR Tier 2, Section 15.0.3.10 states that the spent fuel cask handling equipment is designed so that no cask rupture can occur if a cask is dropped, because the cask cannot be dropped from more than the equivalent of 9.14 m (30 ft). Therefore, no radioactivity is released from a cask drop and the applicant did not perform a radiological consequences analysis of a cask drop. SRP Section 15.7.5 guidance states that no radiological consequences analysis is required if the cask drop height is less than 9.14 m (30 ft). Therefore, the applicant's discussion of the cask drop is acceptable.

For the FHA, the applicant assumed that a single fuel assembly that has undergone 34 hours of decay is dropped, such that the activity in the gap of every rod in the dropped assembly is released. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods (8 percent of I-131, 10 percent of Kr-85, and 5 percent of other iodine and noble gas inventories in the fuel rod) is assumed to occur, with the released gases bubbling up through the fuel pool water. These gap fractions conform to RG 1.183 guidance, and are acceptable for use in the applicant's FHA analysis, because the fuel burnup and linear heat generation rate limitations in the footnote to RG 1.183 Table 3, "Non-LOCA Fraction of Fission Product Inventory in Gap," are met for the U.S. EPR fuel. The applicant assumed an effective decontamination factor of 200 for total iodine as it rises through the fuel pool water. The fuel pool water depth above the fuel is in excess of 7.01 m (23 ft); therefore, the decontamination factor of 200 is acceptable, in accordance with the guidance in RG 1.183. The applicant assumed that iodine in the particulate form is not volatile and is not released. In accordance with RG 1.183 guidance, the applicant assumed that the particulate CsI is converted instantaneously to the elemental form of iodine when it is released from the fuel into the low-pH pool water. The analysis assumed a coincident LOOP.

The staff has reviewed the applicant's analysis and finds that the methods used for the radiological consequence assessment conform to RG 1.183, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant's assessments, the staff performed independent radiological consequence calculations for the FHA occurring 34 hours after shutdown. The staff's analyses followed guidance in RG 1.183 and used the applicant's assumptions on accident progression, fission product source terms, and design reference atmospheric dispersion factors. The offsite radiological consequences calculated by the staff conform to those calculated by the applicant, and confirm that the doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the FHA.

Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the applicant's analysis of the design basis FHA is acceptable. The staff concludes that the U.S. EPR design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated FHA at 34 hours after shutdown will fall well within the dose criterion set forth in 10 CFR 50.47 (i.e., 25 percent or 0.063 Sv (6.3 rem) TEDE).

The applicant reported the results of its radiological consequence analysis for personnel in the main control room and TSC during a design basis FHA, which relies on the CRACS to limit the radioactivity to which the personnel may be exposed. To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the FHA, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The control room and TSC radiological consequences calculated by the staff conform to those calculated by the applicant and confirm that the FHA doses calculated by the applicant are less than 0.05 Sv (5 rem) TEDE.

Therefore, the staff has determined that there is reasonable assurance that the CRACS can mitigate the dose in the main control room and TSC following a design basis FHA to meet the dose criterion given in GDC 19.

15.0.3.4.10 Radiological Consequences of LOCAs

In FSAR Tier 2, Section 15.0.3.11, "Loss of Coolant Accident," the applicant analyzed a hypothetical design basis LOCA for radiological consequences. Additional description of the large break LOCA was given in FSAR Tier 2, Section 15.6.5, "Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary." The applicant concluded that certain bounding sets of atmospheric relative concentration values (χ/Qs) given in FSAR Tier 2, Section 2.3, "Meteorology," in conjunction with the use of natural deposition of fission product aerosol within the containment and the control of the pH of the water in the containment to prevent iodine evolution, are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated design basis LOCA will fall within the relevant dose criteria established in 10 CFR 50.47 and in GDC 19.

The design basis LOCA analyzed for radiological consequences is a postulated accident that results from primary coolant loss in excess of the RCS makeup capacity, thereby leading to meltdown of the core. The DBA LOCA radiological consequences are bounding for the range of RCS break sizes, including small break LOCAs. The DBA LOCA is analyzed to determine if the primary containment is sufficiently capable to prevent release of fission products to the outside environment and also determine if the fission product mitigation systems and features are sufficient. The applicant followed the guidance in RG 1.183 in performing the LOCA radiological consequences analysis.

The U.S. EPR is applying to be licensed with leak-before-break methodology; however, the applicant's analysis did not take credit for the gap release phase onset time of 10 minutes, which would be allowable per RG 1.183 for plants licensed with leak-before-break. The LOCA analysis assumed a coincident LOOP.

The applicant used the core radionuclide inventory discussed above in Section 15.0.3.1 of this report, which assumes operation at 100.5 percent of rated thermal power or 4,612 MWt. In conformance with the guidance in RG 1.183, the analysis considered releases through two pathways; primary containment leakage and engineered safety features (ESF) component leakage.

Containment Leakage Pathway

All releases from the core and RCS are assumed to mix homogenously in the primary containment atmosphere. At the onset of the LOCA, the containment purge system is assumed to be in operation, and the purge flow continues until the primary containment is isolated at 10 seconds. No credit was taken for filtration of the purge exhaust. After isolation, release of the containment atmosphere to the outside environment is through containment leakage based on the proposed TS 5.5.15, "Containment Leakage Rate Testing Program," limit of 0.25 percent per day for the first 24 hours and assuming half that value afterward, until the end of the accident at 30 days. No holdup was credited in the secondary buildings. During the annulus drawdown and safeguard and fuel buildings drawdown for 305 seconds after onset of the accident, 100 percent of the primary containment leakage was assumed to be released directly to the environment as a ground-level release. Following the completion of drawdown, the primary containment leakage is filtered at 99 percent efficiency for all iodine species and released through the plant stack as a ground-level release. No credit is taken for mixing in the shield building annulus volume.

The applicant's analysis takes credit for aerosol natural deposition in the containment based on a combination of the Powers and Henry models incorporated into RADTRAD, with a transition between the models at 22.2 hours. The aerosol deposition was assumed to continue at diminishing rates, for the duration of the accident. Removal of elemental iodine in the containment was assumed to be at the same rate as the aerosol deposition, but ended at about 82 hours when a total decontamination factor of 100 was reached. Organic iodine is not depleted by these processes.

The staff issued the following questions related to aerosol and elemental iodine removal:

In the LOCA and REA dose analyses, the Powers natural deposition model and the Henry model for aerosol removal in containment were combined to model aerosol deposition in the containment. The Henry model has not previously been reviewed or found acceptable for use in reactor licensing. In RAI 17, Question 15.00.03-24, the staff requested that the applicant justify the use of the Henry model for aerosol natural deposition in the EPR containment.

In the LOCA and REA dose analyses, aerosol deposition in containment was modeled in part by using the 10th percentile Powers natural deposition correlation in RADTRAD. The staff noted that the Powers natural deposition correlation was developed using operating pressurized water reactor and boiling water reactor information on containment geometry and power. In RAI 17, Question 15.00.03-25, the staff requested that the applicant justify the applicability of the Powers natural deposition correlation to the EPR containment.

In RAI 17, Question 15.00.03-26, the staff requested that, in regard to the LOCA and REA dose analyses, the applicant provide the methodology for combining the Powers model and the Henry model to give the effective aerosol removal coefficients in FSAR Table 15.0 52 and 15.0 41. The staff further requested that the applicant include the justification for using these models in combination.

In RAI 17, Question 15.00.03-27, the staff further requested that the applicant justify use of models and assumptions on elemental iodine removal, provide the methodology used for elemental iodine removal, assure that this methodology been previously approved for reactor licensing, identify the cases where elemental iodine removal is assumed, and provide the time after the onset of the LOCA that the DF equals 100.

In a September 19, 2008, response to RAI 17, Questions 15.00.03-24 through 27, the applicant provided further explanation and justification of the modeling of aerosol and elemental iodine removal in the U.S. EPR containment. The applicant modeled the aerosol deposition in containment by natural processes by using the tenth percentile correlation from the Powers natural deposition model given in NUREG/CR-6189, "A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments," as implemented in the RADTRAD code for the first 22.2 hours, after which time the Powers model does not provide information. Starting at 22.2 hours, the applicant's analysis used the Henry's aerosol deposition correlation incorporated in the RADTRAD code, given in the Industry Degraded Core Rulemaking (IDCOR) Technical Report 11.7, "FAI Aerosol Correlation."

To justify the use of the Powers aerosol deposition model in RADRAD to model aerosol deposition in the U.S. EPR containment, the applicant provided a comparison of the U.S. EPR to the information used to develop the Powers aerosol natural deposition correlation. The Powers model in NUREG/CR-6189 gives sets of aerosol deposition coefficients over time for several uncertainties, correlated to core thermal power. The Powers correlations were developed based on information on containment dimensions and core thermal power for currently operating light water reactors. The applicant provided a comparison of the ratio of reactor power to the containment volume for the U.S. EPR to several PWRs and showed that the ratio was in the range of the reactors considered in the Powers model. The applicant further showed that the U.S. EPR is comparable in terms of containment size and type to typical currently operating reactors in the U.S. The staff finds the applicant's discussion reasonably justifies application of the Powers aerosol natural deposition model to the U.S. EPR.

To give a smooth transition at 22.2 hours, the applicant back-calculated the Henry's model effective sedimentation height to result in an aerosol deposition coefficient and aerosol density that matches the Powers 10th percentile values at 22 hours. This established inputs for the Henry's model in RADTRAD to give aerosol deposition results with similar conservatism to the Powers 10th percentile correlation. In its RAI response, the applicant also provided information to show that the aerosol deposition coefficient values calculated with their assumptions and the Henry's model in RADTRAD were less than (i.e., more conservative) than those that would have been calculated by the tenth percentile Power correlation if its calculation would have been extended past 22.2 hours. The staff notes that the Henry's model correlation is acceptable in principle if applied to calculating airborne aerosol concentration under dry conditions, since it is based on a number of experiments. Past 22 hours into the accident, the expected airborne aerosol concentration is low; therefore, the importance of the accuracy of the model applied to calculate deposition rate is less important. Therefore, the staff finds that the use of Henry's model as an extrapolation of the Powers model beyond its limits to model aerosol deposition in the U.S. EPR containment is reasonable.

In its September 19, 2008, response to RAI 17, Question 15.00.03-27, the applicant justified the modeling of elemental iodine removal in containment. The applicant stated that elemental iodine will more readily absorb onto the surfaces of the aerosol in lieu of the containment wetted surface as part of their justification for assuming that the aerosol deposition coefficient can be used to model the elemental iodine removal. The staff notes that this argument is not supported by experimental or theoretical evidence; therefore, the staff disagrees that the elemental iodine is preferentially absorbing on the aerosol and does not find the argument persuasive. However, considering the remainder of the information and guidance in SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," on elemental iodine deposition rate, the staff considers the use of the aerosol natural deposition rate as a surrogate for the elemental iodine deposition rate to be reasonably conservative and acceptable.

Based on the applicant's responses to RAI 17, Questions 15.00.03-24 through 27, the staff finds that the applicant has sufficiently addressed the staff's questions on modeling of aerosol and elemental iodine removal in containment.

ESF component leakage pathway

ESF systems that recirculate sump water outside of the primary containment are assumed to leak during their intended operation. RG 1.183, Appendix A guidance on the source term assumptions for the ESF system leakage states that the same source term that was released to the containment in calculating the release through the containment leakage pathway should be assumed to be instantaneously and homogeneously mixed in the sump water as it is released from the fuel. RG 1.183 then states that in lieu of this deterministic approach, suitably conservative mechanistic models for the transport of the airborne activity in the containment to the sump may be used. The applicant chose to use a more mechanistic model that gave a time-dependent IRWST activity source term.

The applicant's analysis assumed that the ESF component total leakage is 0.9 m^3/min (4 gpm), which is twice the proposed limit, in accordance with the guidance in RG 1.183. A 10 percent flashing fraction was applied to model the amount of iodine in the leaked fluid that becomes airborne and released to the environment. During the safeguard and fuel buildings drawdown period for the 305 seconds after onset of the accident, 100 percent of the airborne release from the ESF systems was assumed to be released directly to the environment as a ground-level

release. Following the completion of drawdown, the ESF component leakage is filtered at 99 percent efficiency for all iodine species and released through the plant stack as a ground-level release. No credit for mixing in the Safeguard Building or Fuel Building volumes was taken.

The staff issued a question related to the applicant's ESF leakage pathway assumptions. In RAI 17, Question 15.00.03-30, the staff requested that the applicant provide the ESF component leakage source term and the basis for the assumed values In regard to the LOCA engineered safeguard features component leakage dose analysis. In a September 19, 2008, response to RAI 17, Question 15.00.03-30, the applicant provided the details of the ESF component leakage source term. The time-dependent activity in the IRWST was based on the amount of iodine removed from the containment atmosphere as a result of natural deposition. Containment leakage was modeled as described above for the containment leakage pathway, and the iodine activities were corrected for radioactive decay. The iodine was assumed to immediately enter the IWRST fluid as it was removed from the containment atmosphere, at the natural deposition rate modeled by the Powers and Henry's models implemented in RADTRAD. The applicant's time dependent post-LOCA I-131 activity distribution was given in Table 15.00.03-30-1 of a September 19, 2008, response to RAI 17, Question 15.00.03-30. The staff compared the iodine activity values airborne in containment and IRWST to the results of its own confirmatory calculation of the containment leakage pathway, which modeled aerosol and elemental iodine removal in containment. The staff determined that the total iodine activity and containment airborne iodine values compared well to the values given as output by its confirmatory calculation using RADTRAD, over the 82-hour period modeled. The staff's calculation did not model the IRWST, but the difference between the total and the airborne values give the IRWST values. The staff also performed an independent calculation of the LOCA ESF leakage pathway release, with the RG 1.183 non-mechanistic assumption that the same source term that was released to the containment in calculating the release through the containment leakage pathway is instantaneously and homogeneously mixed in the IRWST water as it is released from the fuel. The staff's LOCA ESF leakage calculation gave dose results that, when combined with the staff's confirmatory calculation results for the LOCA containment leakage pathway, are below the reported doses for the U.S. EPR LOCA analysis and meet the regulatory dose criteria. Based on the September 19, 2008, response to RAI 17, Question 15.00.03-30, as supported by staff's analysis, the staff finds that the applicant has sufficiently addressed the staff's questions on modeling of the fission product source term in the IRWST.

The staff has reviewed the applicant's analysis and finds that the methods used for the radiological consequence assessment conform to RG 1.183, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the applicant's assessment, the staff performed independent radiological consequence calculations for the LOCA. The staff's analyses followed guidance in RG 1.183 and used the applicant's assumptions on accident progression, fission product source terms (with the exception of the IRWST source term), and design reference atmospheric dispersion factors. The offsite radiological consequences calculated by the staff conform to those calculated by the applicant, and confirm that the doses calculated by the applicant are within the dose criteria given in SRP Section 15.0.3 for the LOCA.

Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the applicant's analysis of the design basis LOCA is acceptable. The staff concludes that the U.S. EPR design, as bounded by the

atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated LOCA will not exceed the dose criterion set forth in 10 CFR 50.47 (i.e., 0.25 Sv (25 rem) TEDE).

The applicant reported the results of its radiological consequence analysis for personnel in the main control room and TSC during a design basis LOCA, which relies on the CRACS to limit the radioactivity to which the personnel may be exposed. To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for the LOCA, using the applicant's assumptions on control room and TSC design, design reference atmospheric dispersion factors for the control room and TSC receptors, and the same accident assumptions as for the offsite dose analysis. The control room and TSC radiological consequences calculated by the staff conform to those calculated by the applicant and confirm that the LOCA doses calculated by the applicant are less than 0.05 Sv (5 rem) TEDE. Therefore, the staff has determined that there is reasonable assurance that the CRACS can mitigate the dose in the main control room and TSC following a design basis LOCA to meet the dose criterion given in GDC 19.

15.0.3.4.11 Main Control Room and Technical Support Center Radiological Habitability Analysis

The control room envelope includes the main control room and the TSC and is served by the CRACS. FSAR Tier 2, Section 15.0.3.4.1, "Main Control Room/Technical Support Center Modeling," provides a discussion of the main control room and TSC modeling for the radiological consequences analyses. Additional information on the main control room envelope and CRACS design is given in FSAR Tier 2, Sections 6.4, "Habitability Systems," and 9.4, "Air Conditioning, Heating, Cooling and Ventilation Systems." The staff's review of the control room habitability and the control room ventilation systems is discussed in Section 6.4 and 9.4 of this report, respectively. The U.S. EPR TS include a control room envelope habitability program to maintain the systems and control room envelope.

The DBA control room radiological consequences analyses are bounding for the TSC, because the TSC is located within the main control room pressure boundary and is served by the CRACS in the event of a DBA with radiological release. The CRACS provides a slight positive pressure with the main control room area with respect to adjacent areas during normal and accident conditions. For an accident with release of radioactivity, the outside air supply is automatically diverted through the control room emergency filtration system (CREFS) charcoal and HEPA filter trains. The system is actuated by a primary containment isolation signal (PCIS) or by high radiation levels in the air intake ducts. Filtered recirculation is initiated when the control room is isolated in emergency operation. Each DBA radiological consequence analysis assumed the loss of one train of the two redundant charcoal filtrations systems of the main control room for the duration of the accident. A conservative assumption of an unfiltered inleakage rate of 1.4 m^3/min (50 cfm) is used in the DBA radiological consequences analyses. In its control room dose analyses, the applicant adjusted the dose to the control room personnel due to external gamma radiation from airborne activity within the main control room by applying finite-cloud correction, in conformance with the guidance in RG 1.183.

The applicant's control room radiological consequences analyses also considered external dose from the following sources:

- Radiation shine from the external radioactive plumes
- Radiation shine from radioactive material in the Reactor Building
- Radiation shine from radioactive material in systems and components inside or outside the main control room envelope (e.g., main control room heating, ventilation, and air conditioning (HVAC) filters)

Because of the massive concrete structures surrounding the Reactor Building and main control room, the applicant stated that the only potential source of direct shine radiation to the main control room personnel is the radioactive buildup in the main control room charcoal filtration system. The applicant's calculation of the filter shine dose used the control room design dimensions and selected the assumed receptor location to maximize the filter shine dose. The filter shine dose was calculated at a point 1.8 m (6 ft) above the operating floor of the control room, directly below the filter. The filter was assumed to be a point source located 15.24 cm (6 in.) above the approximately 0.5 m (20 in.) thick concrete floor, therefore making the filter source 3.2 m (10.5 ft) above the receptor location, with the concrete floor providing shielding between. The filter source was based on the iodine loading for the LOCA. The filter shine doses are included in the calculated control room TEDE for each of the DBAs. There is no specific guidance on performing direct dose analyses in RG 1.183. However, the staff has determined that the applicant has used direct dose analysis best practices to model the control room dimensions, source and receptor locations, and used a bounding filter radioactive material source. Therefore, the staff finds that the control room direct dose analysis is acceptable.

During the emergency mode of operation for the control room ventilation system, there are two assumed locations for air intake to the control room envelope: the intake flow through the ventilation system and the unfiltered in-leakage flow. The applicant has calculated a set of reference χ/Q values for each intake location for the control room envelope. Because the computer code (RADTRAD) that the applicant used to perform the DBA radiological consequences analyses cannot accommodate more than one set of atmospheric dispersion factors for the control room dose location, the applicant calculated effective atmospheric dispersion factors and associated main control room intake filter bypass fractions. The staff performed confirmatory calculations to verify the correct calculation of the applicant's effective atmospheric dispersion factors and associated filter bypass fractions. In addition, the staff's independent confirmatory calculations for each of the DBAs described above, did not use the applicant's effective atmospheric dispersion factors and bypass fractions. Instead the staff performed each calculation by separately running a RADTRAD case for each control room intake location/release location pairing, accounting for the total release and control room intake. The staff's calculations further confirmed the applicant's proper use of effective χ/Q values and filter bypass rates. The staff notes that the effective χ/Q values and filter bypass rates used by the applicant in its analysis are intended to ease use of the dose analysis code, and are not site parameter values. The control room and TSC χ/Q values given in FSAR Tier 2, Table 2.1-1 are the values the staff used in its independent calculations and are part of site design envelope.

Based on the comparison of the applicant's analysis methodology to the guidance in RG 1.183 and the staff's confirmatory analysis, the staff finds that the applicant's modeling of the control room and TSC is acceptable, as used in the DBA radiological consequences analyses.

As discussed in the report sections above, the staff has determined that there is reasonable assurance that the main control room design and CRACS can mitigate the dose in the main control room following a design basis accident to meet the dose criterion given in GDC 19.

10 CFR Part 50, Appendix E, Paragraph IV.E.8 requires that an onsite TSC be provided from which effective direction can be given and effective control can be exercised during an emergency. The radiation protection design of the TSC is acceptable if the total calculated radiological consequences for the postulated fission product release fall within the exposure acceptance criteria given for the control room of 0.05 Sv (5 rem) TEDE for the duration of the accident. As discussed in the report sections above, the staff has determined that there is reasonable assurance that the TSC design and CRACS can mitigate the dose in the TSC following a design basis accident to within 0.05 Sv (5 rem) TEDE.

15.0.3.5 *Combined License Information Items*

There are no post-COL activities related to this section.

15.0.3.6 *Conclusions*

The staff has reviewed the radiological consequences analyses of the DBAs described in FSAR Tier 2, Chapter 15, for the U.S. EPR design. Based on the evaluation discussed above, and with the exception of the open items identified above, the staff concludes with reasonable assurance that the U.S. EPR design meets the 10 CFR 52.47(a)(2)(iv) dose criteria and the accident-specific offsite dose acceptance criteria, given in RG 1.183 and the guidance provided in SRP Section 15.0.3.

The staff finds reasonable assurance that the main control room habitability systems, as described in FSAR Tier 2, Section 6.4, can mitigate the dose in the main control room following DBAs to meet the dose criterion given in GDC 19.

The staff finds it reasonable that the main control room habitability systems can mitigate the dose in the TSC following DBAs to be within 0.05 Sv (5 rem) TEDE, to meet the TSC habitability requirements in 10 CFR Part 50, Appendix E, Paragraph IV.E.8 and 10 CFR 50.47(b)(8) and (b)(11).

The staff concludes that, and with the exception of the open items identified above, the information contained in the FSAR Tier 2, Section 15.0.3 conforms to the guidance of RG 1.183 regarding control of the radioiodines in the post LOCA environment as related to the source term iodine assumptions for use in DBA radiological consequences analyses. Therefore, the staff finds the information contained in FSAR Tier 2, Section 15.0.3.12 acceptable.

15.1 Increase in Heat Removal by the Secondary System

Both AOOs and postulated accidents can result in an unplanned increase in heat removal by the secondary system. In these events, a decrease in reactor coolant temperature causes an increase in core reactivity that leads to an increase in core power. The events are described in the following FSAR Tier 2 sections:

- FSAR Tier 2, Section 15.1.1, “Decrease in Feedwater Temperature”
- FSAR Tier 2, Section 15.1.2, “Increase in Feedwater Flow”
- FSAR Tier 2, Section 15.1.3, “Increase in Steam Flow”
- FSAR Tier 2, Section 15.1.4, “Inadvertent Opening of a Steam Generator Relief or Safety Valve”
- FSAR Tier 2, Section 15.1.5, “Steam System Piping Failures Inside and Outside of Containment (PWR)”

Transients and accidents are discussed in FSAR Tier 2, Section 15.0.1. The computer codes used to analyze DBAs are discussed in FSAR Tier 2, Section 15.0.2. Radiological consequences for DBAs are discussed in FSAR Tier 2, Section 15.0.3.

15.1.1 Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve

This section documents the staff’s review of FSAR Tier 2, Sections 15.1.1, “Decrease in Feedwater Temperature,” through 15.1.4, “Inadvertent Opening of a Steam Generator Relief or Safety Valve.”

15.1.1.1 *Introduction*

The decrease in feedwater temperature, increase in feedwater flow, increase in steam flow, and inadvertent opening of a SG relief or safety valve events are all classified as AOOs that are expected to occur with moderate frequency. This classification is in conformance with the guidance in RG 1.206 and NUREG-0800. These events exhibit similar behavior. There is one SRP for these four AOOs, since the regulatory bases and acceptance criteria are identical. These events are discussed as a set below. The steam piping failure accident is discussed separately.

15.1.1.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.1.1 through Section 15.1.4, summarized here, in part, as follows:

15.1.1.2.1 Decrease in Feedwater Temperature

The FSAR states that a decrease in feedwater temperature is postulated to occur due to the inadvertent opening of a feedwater heater bypass valve. The resulting bypass of either a low-pressure or high-pressure feedwater heater division decreases feedwater temperature, thereby increasing heat transfer across the SGs and heat removal from the RCS, lowering the temperature of the RCS. With a negative moderator temperature coefficient, the positive reactivity insertion results in an increase in reactor power potentially leading to a RT. The unplanned power increase associated with this event potentially challenges SAFDLs.

15.1.1.2.2 Increase in Feedwater Flow

An increase in feedwater flow is assumed to be a result of a failure or misoperation of the main feedwater (MFW) control system. The most severe event is a rapid and full opening of a MFW line control valve.

The resulting feedwater flow to the SG causes a decrease in the cold-leg reactor coolant temperature associated with the affected SG. This decrease in temperature causes the automatic ACT control system to withdraw control rods causing an unplanned power increase.

The increase in MFW flow also reduces steam production in the affected SG resulting in a decrease in secondary pressure as the turbine control valve automatically opens to maintain a constant turbine demand. The decreasing secondary pressure increases overall heat transfer to the SGs. A quasi-steady-state condition is maintained until the high SG level trip signal is reached and the MFW system is isolated.

The increase in feedwater flow event potentially challenges SAFDLs. This event can also result in an overfill of an SG, which could force liquid into the steamlines. Therefore, the event is analyzed to assess SAFDLs, as well as SG levels as described in Section 15.1.1.4, "Technical Evaluation," of this report.

15.1.1.2.3 Increase in Steam Flow

The FSAR states that an increase in steam flow can occur when the main steam flow is increased above the steady-state demand flow due to the opening of the turbine control valves or the simultaneous opening of the turbine bypass valves. The unplanned power increase or the rapid decrease in steamline pressure associated with this event can lead to a RT. The increase in power potentially challenges the MDNBR SAFDL.

15.1.1.2.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

The FSAR states that the inadvertent opening of either the SG relief or safety valve can result in a cooldown of the RCS due to more energy being removed from the secondary system than is being generated by the reactor. With a negative moderator temperature coefficient, the decrease in reactor coolant temperature results in an increase in reactor power. The increase in power potentially challenges the MDNBR SAFDL.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications related to FSAR Tier 2 are provided in FSAR Tier 2, Chapter 16, "Technical Specifications," FSAR Tier 2, Sections 3.2.3, "Departure From Nucleate Boiling Ratio (DNBR)," and 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling Limits."

15.1.1.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Sections 15.1.1, "Decrease in Feedwater Temperature," 15.1.2, "Increase in Feedwater Flow," 15.1.3, "Increase in Steam Flow," and 15.1.4, "Inadvertent Opening of a Steam Generator Relief or Safety Valve," and are summarized

below. Review interfaces with other SRP sections also can be found in NUREG-0800, Sections 15.1.1-15.1.4.

1. GDC 10, as it relates to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations including AOOs.

4. GDC 20, as it relates to the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs.

5. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded, including AOOs. This is accomplished by ensuring that the analysis accounts for appropriate margin for malfunctions such as stuck rods.

Acceptance criteria adequate to meet the above requirements include:

1. To identify, which of the moderate frequency initiating events that result in increased heat removal are the most limiting.

2. To verify that, for the most limiting initiating events, the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.

3. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

4. Fuel cladding integrity shall be maintained by ensuring that the MDNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see NUREG-0800, Section 4.4, “Thermal and Hydraulic Design”).

5. To meet the requirements of GDC 10, GDC 13, GDC 15, GDC 20, and GDC 26, the positions of RG 1.105, “Instrument Spans and Setpoints,” are used with regard to their impact on the plant response to the type of transient addressed in NUREG-0800.

6. The most limiting plant systems single failure, as defined in the “Definitions and Explanations” of 10 CFR Part 50, Appendix A shall be identified and assumed in the analysis and shall satisfy the positions of RG 1.53, “Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems.”

7. The analyses of transients caused by excessive heat removal are performed using an acceptable analytical model, and approved methodologies and computer codes. The values of the parameters used in the analytical model should be suitably conservative.

15.1.1.4 *Technical Evaluation*

The events involving an increase in heat removal by the secondary system were analyzed as a group, since they have similar event characteristics, acceptance criteria, and a common NUREG-0800 section. Aspects of methods, models, analysis assumptions, and single failures, which are common to all the events are discussed below followed by issues related to specific events.

The staff confirmed that the analysis of the decrease in feedwater temperature, increase in feedwater flow, increase in steam flow, and inadvertent opening of a SG relief or safety valve events use the approved non-LOCA analytical methodology described in the Codes and Methods Applicability Report for the AREVA TR ANP-10263P-A and the S-RELAP5 computer code to simulate the event. AREVA TR ANP-10287P documents the analytical methodology used to determine the setpoints for the in-core based DNB and LPD LSSS, limitation, and LCO functions in the U.S. EPR. The staff confirmed that the analyses of these events are performed conservatively with most adverse reactivity coefficients, control rod worths, initial power and temperature. The staff notes that the most negative moderator temperature coefficient realized at hot full power is assumed to maximize the reactor kinetic feedback and, therefore, power increase to minimize calculated DNBR, and that minimum flow and maximum heat generation and temperature all minimize DNBR and provide a conservative analysis. No credit is taken for operator actions.

DNB and LPD are computed online. Therefore, establishment of design power peaking limits is not required for the evaluation of thermal margins. Analyses and setpoints are based on pre-computed range of all physically realizable radial and axial power distributions, as explained by the applicant in ANP-10287P.

The FSAR states that both the LOOP and offsite power-available conditions are considered for each event. In conformance with the guidance in RG 1.206 and NUREG-0800, LOOP is not considered the limiting single failure, and an additional failure is postulated in the transient and accident analysis. However, only the worst condition is shown in the FSAR.

FSAR Tier 2, Section 15.2 shows that the RT and subsequent TT are assumed to result in an instantaneous LOOP, and consequential coastdown of the reactor coolant pumps (RCPs). The closing of the turbine stop valves (TSVs) causes SG secondary pressure to increase to the MSRT setpoint, which reduces heat removal from the RCS. The staff observes that these conditions minimize heat transfer from the primary to the secondary side of the SGs, maximize primary and secondary side pressures and temperatures, and are conservative.

FSAR Tier 2, Section 15.1 states that a TT is generated by the check-back signal on a RT. This signal closes the turbine control and stop valves, terminating steam flow to the turbine. Crediting these non-safety-related backup protective systems and components in the DBA analysis following an RT conforms to the regulatory guidance in NUREG-0138. The staff notes that credit for the checkback signal has been a staff position for several decades and is acceptable because of the reliability of the turbine control system.

The staff notes that the U.S. EPR is a four division system, and single failures of one protection system or engineered safety feature division do not affect the functionality of other divisions. The applicant states that the accident analyses presented in FSAR Tier 2, Chapter 15 incorporate the most limiting active single failure of a safety-related system. FSAR Tier 2,

Table 15.0-11 lists the most limiting single failure for each event. Passive failures are not considered, except as event initiators, during the first 24 hours of the event.

To ensure that the limiting single failure had been identified, in RAI 34, Question 15-4, the staff requested that the applicant provide technical bases to demonstrate that the limiting single failure has been selected. In an October 31, 2008, response to RAI 34, Question 15-4, the applicant agreed to modify the FSAR to include a discussion of the technical basis in order to demonstrate that the limiting single failure is selected. The staff reviewed the proposed changes to the FSAR and found them to be acceptable. The staff confirmed that the proposed changes were made in Revision 1 of the FSAR. The staff reviewed the justification for each identified single failure in each postulated transient and accident shown in FSAR Tier 2, Revision 1, Table 15.0.11, "Single Failures Assumed in the Accident Analysis," and found the information acceptable. Therefore, the staff considers RAI 34, Question 15-4 resolved.

Specific Events

Decrease in Feedwater Temperature

The applicant has identified a limiting decrease in feedwater temperature by assuming a 55.5 °C (100 °F) decrease in feedwater temperature, which bounds the mechanistic loss of a single feedwater heater.

In RAI 34, Question 15.01.01-15.01.04-3, the staff requested that the applicant provide bases for the assumed step change in feedwater temperature. In a September 2, 2008, response to RAI 34, Question 15.01.01-15.01.04-3, the applicant explained that a typical winter temperature drop due to the loss of the feedwater heater divisions at HFP is 36.1 °C (65 °F) (27.8 °C (50 °F) plus 8.33 °C (15 °F) uncertainty). An instantaneous drop of 55.5 °C (100 °F) is assumed for this event for additional conservatism. The staff concurs that the assumed decrease in feedwater temperature is conservative. Therefore, the staff considers RAI 34, Question 15.01.01-15.01.04-3 resolved.

The staff was concerned that the limiting flow was used in the analysis. In a September 2, 2008, response to RAI 34, Question 15.01.01-15.01.04-4, the applicant explained that the assumed Technical Specification minimum RCS flow rate used in the analysis is conservative, because it results in a lower MDNBR. The minimum flow also further increases the impact of the overcooling. The staff agrees with the applicant's assessment and concludes that the analysis is conservative. Therefore, the staff considers RAI 34, Question 15.01.01-15.01.04-4 resolved.

In RAI 34, Question 15.01.01-15.01.04-5, the staff noted that FSAR Tier 2, Figure 15.1-7, "Decrease in Feedwater Temperature - RCS and Secondary System Pressures," shows the SG pressure exceeding the main steam relief isolation valve (MSRIV) setpoint of 9,542 kPa (1,384 psia). The FSAR does not explain why the MSRTs did not actuate. If the MSRTs did actuate, it would result in all 4 SGs blowing down until the main steam relief control valves (MSRCVs) close. In an October 31, 2008, response to RAI 34, Question 15.01.01-15.01.04-5, the applicant explained that the analysis was performed with the MSRT pressure control setpoints set at a high pressure of 9,754 kPa (1,414.7 psia) (nominal setpoint of 9,545 kPa (1,384.7 psia) plus an uncertainty of 207 kPa (30.0 psi)). The higher MSRT setpoint did not allow the MSRTs to actuate for the analysis presented in the FSAR.

An additional scenario was analyzed setting the MSRT pressure control setpoint to a low pressure of 9,542 kPa (1,354.7 psia) (nominal setpoint pressure of 9,545 kPa (1,384.7 psia) minus an uncertainty of 207 kPa (30.0 psi)). For this scenario, the MSRTs do actuate, resulting in all four SGs blowing down until the MSRCVs close on low SG pressure. Because the opening and closing of the MSRTs occur after the low DNBR trip at 15.59 seconds, there is no effect on the reactor power response between the two cases. The analysis case described in response to the RAI results in a lower pressurizer pressure, lower primary cold leg temperatures, and a lower pressurizer level than for the FSAR case. The additional analysis shows no change in safety margin from the FSAR case, because the MSRT actuation occurs after the low DNBR trip. Based on this response, the staff concludes that the analysis was conservatively performed since it results in the minimum DNBR. The response is acceptable in that it confirms that the most limiting case has been identified and shown in the FSAR. Therefore, the staff considers RAI 34, Question 15.01.01-15.01.04-5 resolved.

FSAR Tier 2, Figure 15.1-4, "Decrease in Feedwater Temperature - Cold Leg Temperatures," shows asymmetric RCS loop temperatures. In RAI 15.01.01 - 15.01.04-6, the staff requested that the applicant provide an explanation of the asymmetric temperatures. In an October 31, 2008, response to RAI 34, Question 15.01.01-15.01.04-6, the applicant explained that the asymmetric cold leg response of Loop 3 Decrease in Feedwater Temperature - Cold Leg Temperatures is a result of the pressurizer being located on one of the loops and the loss of mixing after the RCPs trip. As the RCS volume shrinks due to the temperature decrease in feedwater temperature, there is an out-surge of higher temperature water from the pressurizer between 20 seconds and 60 seconds. The resulting temperature increase in the Loop 3 hot leg is sufficient to keep the cold leg temperature exiting the Loop 3 SG elevated above the other three loops. The response is acceptable, since it explains the system dynamics and validity of the system modeling. Therefore, the staff considers RAI 34, Question 15.01.01-15.01.04-6 resolved.

Increase in Feedwater Flow

The applicant has identified a full open failure of a MFW control valve as causing the maximum instantaneous increase in feedwater flow. The analysis is performed using the most negative moderator temperature coefficient at hot full power to maximize the reactor kinetic feedback and, therefore, power increase. If the event occurs early in the fuel cycle when the moderator temperature coefficient (MTC) is less negative, the response to a cooldown of a cold leg temperature will cause the automatic ACT control system to withdraw control rods increasing power. In either case, the reactor power increases.

In evaluating this event, the staff recognized that the asymmetry of this event is increased if an increase in feedwater flow to one generator is accompanied by a decrease in feedwater flow to the other SGs. The FSAR analysis is based on a 150 percent increase in feedwater flow to one SG with no decrease in feedwater flow to the remaining SGs. In RAI 34, Question 15.01.01-15.01.04-8, the staff requested that the applicant provide additional information to explain why the event analyzed in the FSAR was limiting. In a September 24, 2008, response to RAI 34, Question 15.01.01-15.01.04-8, the applicant stated:

> For the increase in feedwater flow event, the main feedwater flow is instantaneously increased to a conservative value to a single SG. Total nominal MFW flow is 657 kg/sec (1,449 lbm/sec) per SG. Maximum MFW flow with valves wide open is 689 kg/sec (1,519 lbm/sec) per SG. This means that the

> MFW system is capable of supplying an additional 126 kg/sec (278 lbm/sec) of flow above the nominal value. If all of the additional capacity is assumed to go to the faulted SG, the total flow is 783 kg/sec (1,727 lbm/sec). Using a MFW flow rate of 575 kg/sec (1,267 lbm/sec) calculated at HFP and a reduced MFW temperature of 174 °C (346 °F) steady-state conditions, the percentage increase in MFW flow is 150 percent. A 50 percent increase in feedwater flow to a single SG is conservative, because the valves wide open flow rate only allows for a 5 percent increase in feedwater flow above the nominal full power flow rate. Given the conservative 50 percent increase in feedwater flow to one SG in conjunction with a 55.5 °C (100 °F) decrease in feedwater temperature to all four SGs, an additional reduction in feedwater flow to the unaffected SGs, in order to increase the severity of the event, is not considered necessary.

The applicant's response shows the overall conservatism in the analysis and is acceptable. Therefore, the staff considers RAI 34, Question 15.01.01-15.01.04-8 resolved.

The staff notes that the low DNBR trip provided protection from exceeding SAFDLs. A high SG level trip provides protection from SG overfill.

In RAI 34, Question 15.01.01-15.01.04-9, the behavior of the MSRIV was again questioned. Specifically, FSAR Tier 2, Figure 15.1-18, "Increase in Main Feedwater Flow − SG Pressures," shows the SG pressure exceeding the MSRIV setpoint of 9,542 kPa (1,384 psia). The FSAR does not explain why the MSRTs did not actuate. The staff observed that if the MSRTs did actuate, it would result in all four SGs blowing down until the MSRCVs close.

In an October 31, 2008, response to RAI 34, Question 15.01.01-15.01.04-9, the applicant stated that the analysis presented in the FSAR was conservatively performed with the MSRT pressure control setpoints set at a high pressure of 9,754 kPa (1,414.7 psia) (nominal setpoint of 9,542 kPa (1,384.7 psia) plus an uncertainty of 20.7 kPa (30.0 psi)). The applicant explained that an additional scenario was analyzed setting the MSRT pressure control setpoint to a low pressure of 9,337 kPa (1,354.7 psia) (nominal setpoint pressure of 9,542 kPa (1,384.7 psia) minus an uncertainty of 20.7 kPa (30.0 psi)). For this scenario, the MSRTs actuate earlier than the FSAR analysis (195.7 seconds compared to 214 seconds). When the MSRTs actuate (low MSRT setpoint, RAI case) at approximately 195.7 seconds, all four SGs blow down until the main steam relief control valves close on low SG pressure at 234.6 seconds. Because the opening and closing of the MSRTs occur after the high SG level trip at 188.7 seconds, there is no effect on the reactor power response between the two cases. The analysis performed in response to the RAI results in a lower pressurizer pressure, lower primary cold leg temperatures, and a lower pressurizer level than for the U.S. EPR FSAR case. The analysis shows no change in safety margin from the FSAR analysis, because the MSRT actuation occurs after the high SG level trip. The thermal-hydraulic response is asymmetric, since only one SG is affected by the failure of a feedwater control valve. The analysis in the FSAR is shown to be more limiting, and the response is acceptable. Therefore, the staff considers RAI 34, Question 15.01.01-15.01.04-9 resolved.

In RAI 34, Question 15.01.04-10, the staff requested that the applicant clarify what appears to be a sectorized core model. In a September 2, 2008, response to RAI 34, Question 15.01.04-10, the applicant explained that to provide a more realistic reactivity feedback model, a split (or sectorized) core model is used for this event. The split (or sectorized) channel core model was developed by modifying the non-LOCA single channel

core S-RELAP5 model to partition its single-channel core (and the other flow paths within the reactor vessel) into a two-channel arrangement, with no cross-flow or mixing between the affected region (representing one-fourth of the core/reactor vessel cross-section) and the unaffected region (representing the remaining three-fourths of the core/reactor vessel cross-section). The approved single-channel model nodalization has been modified to a split-core using features of the approved steamline break model. The modifications are straightforward and acceptable since they provide a more detailed nodal representation while preserving phenomenological aspects of the model. Therefore, the staff considers RAI 34, Question 15.01.04-10 resolved.

Increase in Steam Flow

The applicant has postulated the opening of the six turbine bypass valves as a representative event in this category. The staff finds this acceptable, since smaller increases in flow will cause less dynamic response, while FSAR Tier 2, Section 15.1.4, and Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment (PWR)," describe more bounding events.

It is assumed that the RT and TT will cause a LOOP that terminates MFW flow and starts the coastdown of the RCPs. That staff notes that the analysis is performed conservatively with limiting values of reactivity coefficients, control rod control and control rod worths, initial power and temperature to minimize calculated DNBR. The increase in secondary system pressure to the MSRT setpoint and the transition to natural circulation causes the heatup and pressurization of the RCS. The PSRVs open to control RCS pressure while overpressure protection is provided on each of the four main steamlines by an MSRT and two main steam safety valves (MSSVs), therefore, meeting primary and secondary system pressure acceptance criteria.

Inadvertent Opening of a Steam Generator Relief or Safety Valve

Each SG has a MSRT consisting of a main steam relief isolation valve in series with a main steam relief control valve. The MSRIV is normally closed. The MSRCV is normally open. It is assumed that the initiating event is the inadvertent opening of the MSRIV, and the worst single failure is failure of the corresponding MSRCV, allowing blowdown of the affected SG.

FSAR Tier 2, Section 15.1 states that the inadvertent opening of a main steam safety valve is bounded by the inadvertent opening of a MSRT, since the relief capacity of a MSRT is 50 percent of full steam load, while the relief capacity of a MSSV is 25 percent of full load. Although the inadvertent opening of a relief valve occurs in one SG, the four SGs pick up and share the extra load within a few seconds, because their steamlines are connected at the turbine bypass header. Consequently, conditions in the RCS remain symmetric until RT and closure of the main steam isolation valves (MSIVs). The RTs on low DNBR and the subsequent TT are assumed to cause a LOOP that terminates MFW flow and starts the coastdown of the RCPs.

The FSAR shows the results of a case at end of cycle (EOC) HFP in which the non-safety-related automatic ACT control system is disabled. The FSAR states that multiple cases were analyzed at beginning of cycle (BOC) and EOC, HFP, 25 percent power, and HZP, and the corresponding reactor kinetics and scram reactivity to identify the most limiting case. Since the sensitivity studies span realizable conditions, the staff accepts these sensitivity studies as identifying the most limiting case.

There is no unique DNBR as a function of time in these events, because DNBR is computed on line using the measured power distribution, temperature, and pressure. In a September 24, 2008, response to RAI 34, Question 15.01.01-15.01.04-1, the applicant provided representative normalized DNBR and fuel temperature or clad strain limits. The staff observes that the events are characterized by an increase in heat removal leading to a decrease in primary system temperature. With a negative MTC, this results in an increase in power, and therefore fuel clad temperature. Primary system pressure may also decline. These conditions result in a decrease in DNBR margin. The plots of DNBR and fuel temperature or clad strain normalized to their limiting value figures show this degradation followed by an immediate increase in margin following RT. The applicant's response conforms to anticipated system behavior and is acceptable. Therefore, the staff considers RAI 134, Question 15.01.01-15.01.04-1 resolved.

In RAI 34, Question 15.01.01-15.01.04-11, the staff requested that the applicant provide clarification of which MSRCV (SG-3 or SG-4) is assumed to be failed open. In a September 24, 2008, response to RAI 34, Question 15.01.04-11, the applicant stated that the initiating event for the "Inadvertent Opening of a Steam Generator Relief or Safety Valve," is an inadvertent opening of the SG Loop 3 main steam relief isolation valve (affected SG) in conjunction with a single failure of the associated main steam relief control valve to close. The analyses of initiating events with a symmetric core configuration (e.g., FSAR Tier 2, Sections 15.1.1, 15.1.3, and 15.1.4) are simulated as occurring in SG Loop 3. The analyses of initiating events with asymmetric core configurations (e.g., FSAR Tier 2, Sections 15.1.2 and 15.1.5) are simulated as occurring in SG Loop 4. The response is acceptable in that it clarifies the dynamic response shown in the analysis figures and confirms the limiting case. Therefore, the staff considers RAI 34, Question 15.01.01-15.01.04-11 resolved.

In a September 2, 2008, response to RAI 34, Questions 15.01.01-15.01.04-2, and 15.01.01-15.01.04-7, editorial corrections and clarifications were provided. In RAI 34, Question 15.01.01-15.01.04-12, the staff requested that the applicant provide clarification of why SG-3 has higher heat transfer (FSAR Tier 2, Figure 15.1-36, "Inadvertent Opening of an SG Relief or Safety Valve – Heat Transfer to SGs"), yet a higher outlet temperature (FSAR Tier 2, Figure 15.1-37, "Inadvertent Opening of a SG Relief or Safety Valve – Cold Leg Temperatures") compared to the other loops. In a September 24, 2008, response to RAI 34, Question 15.01.01-15.01.04-12, the applicant stated that: The slightly higher cold leg response of Loop 3 is a result of the pressurizer being located on this loop and the loss of mixing after the RCPs trip. As the RCS volume shrinks due to the temperature decrease (due to the decrease in feedwater temperature), there is an out surge of higher temperature water from the PZR. The resulting temperature increase in the Loop 3 hot leg is sufficient to keep the cold leg temperature exiting the Loop 3 SG slightly elevated above the other three loops. The response is acceptable, since it explains the system dynamics. Therefore, the staff considers RAI 34, Question 15.01.01-15.01.04-12 resolved.

In RAI 34, Question 15.01.01-15.01.04-13, the staff requested that the applicant provide information to support the conclusion that the peak return to power for each HZP case does not cause fuel damage. In a September 24, 2008, response to RAI 34, Question 15.01.01-15.01.04-13, the applicant stated that: For the HZP cases, the analysis concludes that the MDNBR is 2.77, and the peak LPD for this event is 39.4 kW/m (12.02 kW/ft). The maximum LPD realized for this event is significantly below the LPD limit 56.4 kW/m (17.2 kW/ft) that precludes both 1 percent clad strain and FCM. This supports the conclusion in FSAR Tier 2, Section 15.1.4.3.1, "Inadvertent Opening of an MSRIV," that the peak return to power does not cause fuel damage. The response shows that the return to power does not

result in fuel damage and is, therefore, acceptable. Therefore the staff considers RAI 34, Question 15.01.01-15.01.04-13 resolved.

15.1.1.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.1.1.6 *Conclusions*

The applicant has identified the most limiting event sequence in each of the four events in conformance with the guidance in RG 1.206 and NUREG-0800, Sections 15.1.1-15.1.4.

The analyses show that there is 95 percent probability with 95 percent confidence that the DNBR limit is met.

The analyses show that the primary and secondary system pressure does not exceed 110 percent of the design limits.

The most limiting single failure has been assumed in the analyses which conform to the guidance in RG 1.53. LOOP has been taken as a consequential failure rather than a single failure which conforms to the guidance in RG 1.206 and NUREG-0800, Sections 15.3.1 and 15.3.2.

The systems analyses conform to RG 1.105 in that instrument uncertainties are included in the analytic limit, and comply with GDC 13, in that only safety-related trips are used in the analysis.

The analyses have used input assumptions and methods to minimize calculated DNBR and maximize primary and secondary system pressure. The analyses use approved methodologies and previously approved computer codes. Variation in nodalization was explained and found acceptable.

The events do not cause a more serious event. Sufficient safety-related equipment is available to shut down the reactor.

Therefore, the staff finds that:

1. GDC 10 is met in that the safety analysis shows that SAFDLs are not exceeded during normal operations including AOOs.

2. GDC 13 is met in that the safety analysis takes credit for only safety-related instrumentation to monitor variables and systems over their anticipated ranges to maintain variables and systems within prescribed operating ranges.

3. GDC 15 is met in that the safety analysis shows that the primary system pressure is maintained at less than 110 percent of design and that there is margin to ensure that the pressure boundary will not be breached during normal operations including AOOs.

4. GDC 20 is met in that the accident analysis shows the reactor protection system initiates automatically to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs.

5. GDC 26, is met in that the safety analysis shows that timely rod insertion ensures that SAFDLs are not exceeded, including AOOs assumed including malfunctions such as stuck rods are accounted for.

15.1.2 Increase in Feedwater Flow

Review of this section of the FSAR is documented under Section 15.1.1 of this report.

15.1.3 Increase in Steam Flow

Review of this section of the FSAR is documented under Section 15.1.1 of this report.

15.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

Review of this section of the FSAR is documented under Section 15.1.1 of this report.

15.1.5 Steam System Piping Failures Inside and Outside of Containment (PWR)

15.1.5.1 *Introduction*

A steam system piping failure inside or outside of containment can result in an unplanned increase in the heat removal capability of the secondary system leading to an unplanned increase in core power. In accordance with the classification of events described in FSAR Tier 2, Section 15.0.0.1, minor steam system piping failures are considered AOOs, and larger breaks are treated as postulated accidents (PAs). The analyses described in FSAR Tier 2, Section 15.1.5 address large MSLBs, which are PAs.

15.1.5.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.1.5 of this report summarized here, in part, as follows:

The rupture of a main steam pipe increases the rate of energy removal from the RCS, which lowers RCS temperatures and pressures. When the low SG pressure or high SG pressure decrease setpoint is reached, the protection system initiates a RT and closes the main steam isolation valves.

Positive reactivity is inserted because of the assumed negative MTC. This condition reduces core shutdown margin, which can lead to a return to criticality and power. The SAFDLs, including cladding strain, can be challenged if shutdown margin is eroded sufficiently for the reactor to return to power. Where SAFDLs are exceeded, the fuel is assumed to fail. The resulting fuel failure fractions are used to determine radiological consequences.

Steam system piping failure events are analyzed using the approved non-loss-of-coolant accident analytical methodology described in the Codes and Methods Applicability Report for the AREVA TR ANP-10263P-A and the S-RELAP5 computer code to simulate the event. The LYNXT computer code is used to determine the MDNBR for the event. The PRISM computer

code is used to calculate the core power distributions. A spectrum of break size cases is analyzed for conservative combinations of core power levels, break locations (upstream and downstream of the main steam isolation valve), offsite power conditions, and single failures.

The FSAR analysis concludes that the regulatory acceptance criteria are met and that radiological releases are within the limits of the 10 CFR Part 50, Appendix A and 10 CFR 50.34(a)(1) criteria.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications related to FSAR Tier 2, Section 15.1.5 are provided in FSAR Tier 2, Chapter 16, Section 3.2.3, "Power Distribution Limits, Departure From Nucleate Boiling Ratio."

15.1.5.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.1.5, "Radiological Consequences of Main Steam Line Failures," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.1.5.

1. GDC 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

2. GDC 17, as it relates to the requirement that an onsite and offsite electric power system be provided to permit the functioning of SSCs important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that the acceptable fuel design limits and the design conditions of the RCPB are not exceeded during an AOO and that core cooling, containment integrity, and other vital functions are maintained in the event of an accident.

3. GDC 27 and GDC 28, as they relate to the RCS being designed with appropriate margin to ensure that acceptable fuel design limits are not exceeded and that the capability to cool the core is maintained.

4. GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

5. GDC 35, as it relates to the reactor cooling system and associated auxiliaries being designed to provide abundant emergency core cooling.

6. 10 CFR Part 50, Section 50.34(a)(1), "Contents of applications; technical information," as it relates to the evaluation and analysis of the offsite radiological consequences of PAs with fission product release.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

2. The potential for core damage is evaluated on the basis that it is acceptable if the MDNBR remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see NUREG-0800, Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see NUREG-0800, Section 4.2, "Fuel System Design"), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

3. The radiological criteria used in the evaluation of steam system pipe break accidents (PWRs only) are provided in NUREG-0800, Section 15.0.3.

4. The integrity of the RCPs should be maintained such that loss of alternating current (ac) power and containment isolation will not result in pump seal damage.

5. The auxiliary feedwater system (AFWS) or other means of decay heat removal must be safety-related and, when required, automatically initiated.

6. Tripping of the reactor coolant pumps should be consistent with the resolution of Task Action Plan item II.K.3.5.

15.1.5.4 *Technical Evaluation*

The applicant has applied its approved MSLB methodology for operating PWRs described in TR ANP-10263, "Codes and Methods Applicability Report of the U.S. EPR," to the analysis of the MSLB in the U.S. EPR. The applicant justified the applicability of this methodology to the U.S. EPR in a technical report that was conditionally approved by the staff (Ref. NRC SER dated June 16, 2009). The condition imposed by the staff was that the applicant would justify the SG modeling via sensitivity studies described below.

The staff confirmed that the U.S. EPR accident analysis models four trains of automatically actuated emergency feedwater, that the accident analysis does not postulated loss of RCP cooling upon containment isolation and that the accident analysis considers the automatic trip of the RCPs in the U.S. EPR design. More information is contained in Chapter 5, "Reactor Coolant System and Connected Systems," of this report.

For the MSLB analysis, the SG secondary side is modeled as two volumes. A lower volume represents everything below the steam dome, and an upper volume represents the steam dome. Only steam is allowed to pass from the lower to the upper volume. The applicant has argued that this simplified noding of the SG secondary is conservative and adequate for MSLB analyses (Response to RAI 10 for "Codes and Methods Applicability Report" for the AREVA TR ANP-10263P-A), and the staff concurs, since this model maximizes heat removal from the primary and maximizes the return to power and challenge to the fuel.

As discussed further in the applicant's August 29, 2008, response to RAI 34, Question 15.01.05-6 (below), the condition imposed upon TR ANP-10263P does not apply to the applicability of the applicant's methodology to MSLB analyses.

To better judge the results presented in the FSAR, the staff requested the results of the other MSLB cases. In RAI 34, Question 15.01.05-1, the staff requested that the applicant provide the results of pipe failures inside the containment and provide justification that the 0.16 m^2 (1.72 ft^2) break outside containment is limiting with respect to core performance. In a September 2, 2008, response to RAI 34, Question 15.01.05-1, the applicant stated that the main steam pressure sensors are located outside containment, and a break outside containment would expose them to harsh environmental conditions. The normal and degraded pressure uncertainties are 20.7 kPa (30 psi) and 62.1 kPa (75 psi), respectively. Because of the impact of a break outside containment on the protection system response, a break outside containment generates more limiting results than a break inside containment, because a break outside containment results in a later actuation of the protection system.

In RAI 34, Question 15.01.05-2, the staff requested that the applicant provide the results of HFP and HZP cases with and without LOOP, so that the staff could verify the accident scenario presented is indeed limiting. In a September 24, 2008, response to RAI 34, Question 15.01.05-2, the applicant provided a table of results for the MSLB analyses. A spectrum of break sizes has been considered and so has the initial state of the reactor (HFP, HZP, and partial power). The limiting single failure, failure of a MSRCV to close in one of the unaffected SGs for breaks upstream of the MSIV or failure of the MSIV for breaks further downstream, has been identified and incorporated into the analyses. The MSLB analysis, using approved methods, has considered both LOOP and no LOOP. The results of the analyses show a return to power as high as 23 percent core average power, and a conservative estimate of the amount of failed fuel is 1.24 percent of the total fuel in the reactor. The radiological consequences of the MSLB event are treated in FSAR Tier 2, Section 15.0.3.7, "Main Steam Line Break Outside of Reactor Building Accident."

From the response, it is clear that the number of calculated fuel failures was identical for many cases. This is because the methodology assumes that if any fuel rod in an assembly is calculated to fail, then it is assumed that all rods in that assembly fail. This procedure results in several cases having a 1.24 percent fuel failure even though the precise number of calculated failures differs between the cases. Although many cases had the same number of fuel failures, the applicant chose to present the peak post-scram power case in the FSAR. The procedure conservatively estimates the extent of fuel failure and is acceptable.

The applicant presented an analysis of only the post-scram, from reactor trip through return to power and return to subcriticality, portion of the MSLB in the FSAR. In RAI 34, Question 15.01.05-3, the staff requested that the applicant provide an analysis of the pre-scram, from initiating event to reactor trip, portion of the MSLB be conducted and documented in the FSAR. In a September 24, 2008, response to RAI 34, Question 15.01.05-3, the applicant stated that for the pre-scram MSLB, the potential exists for a delay in RT, because instrument decalibration due to harsh containment conditions results in longer blowdown and greater cooldown potentially challenging SAFDLs (i.e., cladding strain and/or FCM). The applicant explained that although this event is usually dispositioned for existing PWR designs as being bounded by the post-scram MSLB events, it is evaluated for the U.S. EPR based on the potential impact of the anticipated harsh environmental conditions in the sensor area of the various steamline break locations and because of the asymmetric core effects of the post-scram

MSLB event versus the anticipated symmetric effects of the pre-scram MSLB events. A pre-scram MSLB event is considered symmetric because of the absence of main steamline (MSL) check valves in the U.S. EPR design and the presence of a main steam (MS) bypass common header upstream of the turbine stop valves.

Medium-break and double-ended guillotine break analyses are performed with a deterministic LYNXT code calculation. The results show that the pre-scram, at power cases have an MDNBR approximately equal to the DNBR LCO and a maximum calculated LPD of 14.99 kW/ft. Therefore, the pre-scram portion of the MSLB event does not challenge the DNB or LPD SAFDLs. Accordingly, the staff determined that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 34, Question15.01.05-3 resolved.

The staff was concerned that the applicant's analysis had considered a range of initial conditions for its MSLB analyses and had chosen a conservative set of initial conditions for its analyses. In RAI 34, Question 15.01.05-4, the staff requested that the applicant explain how it was determined that intermediate power levels do not present a more challenging initial condition, since the U.S. EPR is designed to have a constant T_{avg} above 60 percent power. In an October 31, 2008, response to RAI 34, Question 15.01.05-4, the applicant stated that to bound this average coolant temperature programming, the spectrum of post RT main steamline break cases included cases initiated at 0 percent power, 25 percent power, 60 percent power, and 100 percent power. The worst case was determined to be initiated at 0 percent power. The staff finds the sensitivity study used to determine that the zero case is limiting acceptable, therefore, the staff considers RAI 34, Question 15.01.05-4 resolved.

In RAI 34, Question 15.01.05-5, the staff requested that the applicant explain why the Technical Specification minimum flow used in the analysis is conservative. In a September 2, 2008, response to RAI 34, Question 15.01.05-5, the applicant stated that the assumed Technical Specification minimum RCS flow rate is conservative, because it results in a lower MDNBR. In addition, the lower flows result in a larger temperature difference across the core and SGs. This causes a decrease in the cold leg coolant temperatures that, in combination with a negative moderator temperature coefficient, maximizes the reactor power increase. This further increases the impact of the overcooling. The staff finds that the applicant has adequately addressed this issue and, therefore, the staff considers RAI 34, Question15.01.05-5 resolved.

In RAI 34, Question 15.01.05-6, the staff requested that the applicant explain why modeling the SG secondary boiler region as a single volume is conservative. In a September 2, 2008, response to RAI 34, Question 15.01.05-6, the applicant stated that this modeling issue had been addressed previously.

The applicant stated that this question is similar to RAI 10 related to AREVA TR ANP-10263P-A. The RAI and the applicant's April 24, 2007, response in TR ANP-10263Q1NP, Revision 1, are as follows:

Page 5-33 of TR ANP-10263NP states: "Steam generators with axial economizers: The U.S. EPR steam generator design features an axial economizer, which channels the feedwater through the 180 degree sector of the downcomer and lower boiler regions that corresponds to the cooler, downflow legs of the U-tubes. However, modeling this feature is not warranted for a Main Steamline Break analysis, in which all of a steam generator's downcomer and boiler regions are lumped together into a single volume and the feedwater is injected into the bottom of that volume."

For the U.S. EPR, main feedwater is terminated early in the MSLB transient, and emergency feedwater is initiated. While main feedwater is injected directly into the top of the economizer, emergency feedwater is sprayed relatively high in the generator above the economizer structure. Both hot and cold sides of the downcomer receive emergency feedwater, thus effectively negating the effect of the economizer. This makes it unnecessary to model the economizer explicitly, since it performs its function for only a very short period during the transient.

Without the economizer, therefore, the generator is equivalent to those in typical Westinghouse or Combustion Engineering designs. The model used to analyze MSLB events in those plants will perform as well for the U.S. EPR.

The MSLB model for the U.S. EPR follows the approved methodology prescribed in TR ANP-10263P-A. The basis of the methodology is that fuel integrity (and primary loop integrity) is dependent on the heat removal rate from the primary. Conservative treatment of the break - allowing steam only out the break - maximizes the heat removal from the primary. This maximizes the reactivity impact of the cooldown, leads to the greatest probability of return to power, and maximizes the effect of the cooldown on the primary side level control system (pressurizer). From this perspective, the SG provides a boundary condition to the primary system.

As prescribed by the MSLB methodology, the SG secondary side fluid volume below the top of the primary steam separators is lumped in a single control volume. The fluid volume above this point is included in a second control volume (the steam dome), connected to the lower region by a junction that allows only steam to pass to the dome. This perfect separation of the liquid and vapor phases results in the removal of more energy from the SG than would occur if a two-phase mixture were allowed to leave the SG through the break (if an explicitly modeled separator were used).

Using a single control volume to represent the lower portion of the SG is conservative for the MSLB analysis. The incoming cold feedwater is mixed with all of the liquid in the SG and produces a lower bulk secondary side fluid temperature than if the feedwater is mixed with only a portion of the SG liquid in a more complex nodalization that does not have the feedwater in contact with all of the fluid.

Sensitivity studies on the nodalization are not required, because adding detailed nodalization tends to impede cooldown associated with the simplified model. The passive heat structures, such as the SG shell wall, the tube wrapper, the tube divider plate, and the economizer shell are conservatively omitted from the model as these would release heat as the SG cools down, thus mitigating the temperature reduction.

It is clear from the applicant's response that the modeling of the SG secondary side in S-RELAP5 is conservative. The staff finds that the applicant's response is acceptable, therefore, the staff considers RAI 34, Question 15.01.05-6 resolved.

In RAI 34, Questions 15.01.05-7 through 15.01.05-11, the staff requested that the applicant provide clarification of information contained in the FSAR or to address editorial problems with the FSAR. In responses dated September 2, 2008, and October 31, 2008, the applicant adequately addressed these issues and, therefore, the staff considers RAI 34, Questions 15.01.05-7 through 15.01.05-11 resolved.

There is a reactor power increase prior to the end of the return to power shown in FSAR Tier 2, Figure 15.1-55, “MSLB - Reactor Power.” The staff requested that the applicant provide an explanation of the increase in RAI 34, Question 15.01.05-12. In a September 24, 2008, response to RAI 34, Question 15.01.05-12, the applicant stated that the power increase is a consequence of a chain of events initiated by the drying out of the affected SG. As the affected SG begins to dry out, the choked flow condition at the SG outlet nozzle terminates, and SG pressure falls rapidly, enhancing boiling heat transfer on the tubes.

The applicant states that while the SG U-tubes are modeled by means of twelve axial nodes, two of which represent the tube sheet region, and 10 heat conductors that provide heat transfer to the SG boiler region, the boiler region is modeled with a single node. In the physical SG, the small affected SG water inventory during the return to power would be in contact only with the tube surfaces near the tube sheet. In the modeling of the boiler region, all SG tube thermal conductors are in contact with the boiler region water inventory.

During the return to power, the great majority of the heat transfer from the reactor coolant in the tubes to the affected SG boiler region is via the first tube thermal conductor. When the SG begins to experience dryout, this heat conductor experiences DNB, and the majority of heat transfer transitions to the next tube heat conductor, which soon also experiences DNB. This initiates a cascading chain of events that ends when the last tube heat conductor experiences DNB.

During the DNB cascade, net SG heat transfer falls off rapidly. Prior to its experiencing DNB, the mass of reactor coolant adjacent to tube conductor No. 1 passes through tube conductor No. 2, and all subsequent tube conductors prior to their experiencing DNB. Therefore, in spite of the net reduction in SG heat transfer, a slug of reactor coolant experiences enhanced heat transfer throughout its transit through the affected SG tubes and is overcooled. When this overcooled slug of water reaches the core, it adds positive reactivity and causes the spike in power.

While this power spike is caused by a brief enhancement of SG tube heat transfer associated with dryout, its magnitude is an artifact of the single-node boiler region in the MSLB model. As much of the cooling of the reactor coolant occurs in tube regions that would be above the water level in the SGs, the single-node boiler region over predicts the cooling of the reactor coolant and the resultant power spike.

The staff accepts the applicant’s explanation of the steam generator heat transfer based on its thermal-hydraulic understanding. Therefore, the staff considers RAI 34, Question 15.01.05-12 resolved.

The MDNBR in the core occurs during the return to power during the accident when temperature, pressure, and flow are low. The staff was concerned about the applicability of DNBR correlations under such situations. In RAI 141, Question 04.04-36, the staff requested that the applicant justify the applicability of its critical heat flux (CHF) correlation for the MSLB with LOOP. The staff questioned the applicability of the AHC-2 and BWU-N CHF correlations to compute DNBR for the MSLB event. In a November 20, 2008, response to RAI 141, Question 04.04-36, the applicant explained that the Biasi CHF correlation, which is included in the NRC-approved LYNXT code, is used for evaluating DNB for the post-scram MSLB with LOOP event rather than the BWU-N or ACH-2 correlations. The applicant explained that the Biasi CHF correlation provides a conservative evaluation of DNB and has a much broader range of applicability for local mass flux, relative to either BWU-N or ACH-2. The use of the Biasi CHF

correlation for evaluating MSLB events conforms to AREVA TR ANP-10263P-A. A table showing the range of applicability of the Biasi correlation was provided. The applicant stated that parameters either fall within the range of applicability of the Biasi CHF correlation or the DNBR is conservatively calculated when the parameters are outside of the range of applicability. The staff accepts the use of the Biasi correlation since the range of applicability of the thermal hydraulic parameters in the correlation span the range of computed values in the safety analysis. Therefore, the staff considers RAI 141, Question 04.04-36 resolved.

15.1.5.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.1.5.6 *Conclusions*

The staff concludes that the analysis for postulated MSLBs is acceptable for the following reasons:

- Modeling techniques and input parameter values have been chosen conservatively to give the maximum cooldown effect and maximum reactivity insertion.
- The number of failed fuel rods is conservatively calculated yet still remains small enough that the core remains intact with no loss of cooling capability.
- The applicable SRP acceptance criteria have been met:
    - Pressures in the RCS and the main steam system remain below 110 percent of the design limits.
    - Fuel failure is of limited extent that the core remains in place and intact with no loss of core cooling capability.
    - The radiological consequences of the MSLB are discussed in FSAR Tier 2, Section 15.0.3.
    - The integrity of the RCPs is maintained in that containment isolation in the MSLB sequence does not cause loss of RCP cooling.
    - The emergency feedwater (EFW) is safety-related and actuated automatically by the safety-related protection system.
    - The most limiting single failure has been assumed in the analysis and conforms to RG 1.53. LOOP has been taken as a consequential failure rather than a single failure in conformance with the guidance in RG 1.206.
    - The systems analysis conforms to RG 1.105, in that instrument uncertainties are included in the analytic limit, and complies with GDC 13, in that only safety-related trips are used in the analysis.
    - The event does not cause a more serious event. Sufficient safety-related equipment is available to shut down the reactor.

Therefore, the staff finds that:

1. GDC 13 is met in that the analysis takes credit for instrumentation to monitor variables and systems and of controls to actuate RT and emergency feedwater.

2. GDC 17 is met in that the analysis considers normal power and loss of offsite power scenarios.

3. GDC 27 and GDC 28 are met in that the analysis considers reactor scram with the most reactive RCCS stuck in its' withdrawn position.

4. GDC 31 is met in that vessel cooldown is not predicted to challenge the RCS boundary behavior.

5. GDC 35 is met in that the analysis assumes one of four trains of SI is failed and that another train is out of service.

6. 10 CFR Part 50, Section 50.34(a)(1), "Contents of applications; technical information," is met as it relates to the extent of fuel failures used in the evaluation and analysis of the offsite radiological consequences of PAs with fission product release.

15.2 Decrease in Heat Removal by the Secondary System

Several AOOs and one postulated accident result in an unplanned decrease in heat removal by the secondary system. These events are described in the following FSAR Tier 2, Section 15.2, "Decrease in Heat Removal by the Secondary System," subsections:

- FSAR Tier 2, Section 15.2.1, "Loss of External Load (LOEL)"
- FSAR Tier 2, Section 15.2.2, "Turbine Trip"
- FSAR Tier 2, Section 15.2.3, "Loss of Condenser Vacuum"
- FSAR Tier 2, Section 15.2.4, "Inadvertent Main Steam Isolation Valve Closure"
- FSAR Tier 2, Section 15.2.5, "Steam Pressure Regulator Failure" (Not applicable to the U.S. EPR)
- FSAR Tier 2, Section 15.2.6, "Loss of Non-Emergency AC Power to the Station Auxiliaries"
- FSAR Tier 2, Section 15.2.7, "Loss of Normal Feedwater Flow"
- FSAR Tier 2, Section 15.2.8, "Feedwater Line Breaks Inside and Outside Containment"

Transients and accidents are discussed in FSAR Tier 2, Section 15.0.0. The computer codes used to analyze DBAs are discussed in FSAR Tier 2, Section 15.0.2. Radiological consequences for DBAs are discussed in FSAR Tier 2, Section 15.0.3.

15.2.1 Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, and Inadvertent Main Steam Isolation Valve Closure

This section documents the staff's review of FSAR Tier 2, Sections 15.2.1 thru 15.2.4.

These events are discussed together, since they all involve a loss of normal heat removal by the secondary side power conversion system. They share a common NUREG-0800 section and common acceptance criteria.

15.2.1.1 *Introduction*

The loss of external load event is an AOO initiated by an electrical disturbance that causes the loss of a significant portion of the turbine generator load resulting in a decrease in heat removal by the secondary system and a corresponding temperature and pressure increase of the RCS.

In a TT event, a malfunction in a turbine or reactor system causes the turbine to trip off line by rapidly stopping steam flow to the turbine resulting in a decrease in heat removal by the secondary systems and a corresponding temperature and pressure increase in the RCS.

A loss of condenser vacuum (LOCV) event is a malfunction that can result in a TT event with a corresponding decrease in heat removal by the secondary system and a corresponding temperature and pressure increase of the RCS. In addition, due to system interaction, the loss of condenser vacuum event also causes the loss of main feedwater to the secondary side of the SGs.

The inadvertent closure of a main steam isolation valve is an AOO resulting from a steamline or reactor system malfunction or inadvertent operator actions.

15.2.1.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.2.1, summarized here, in part, as follows:

The FSAR states that the impact of a loss of external load due to an electrical disturbance can be mitigated by various non-safety functions and equipment without resulting in a RT. However, no credit for these non-safety functions is taken in the analysis. As a result, the FSAR analysis assumes the safety-related MSRTs and main steam safety valves operate to control SG pressure and the safety-related pressurizer safety relief valves (PSRVs) open to control RCS pressure. Following turbine control valve (TCV) closure, secondary side pressure and temperature increase significantly as thermal energy from the RCS continues to be transferred to the SGs. Increasing secondary system temperature causes a corresponding increase in RCS temperature. This increase in reactor coolant temperature causes an expansion of the reactor coolant inventory into the pressurizer and an increase in RCS pressure.

The severity of the LOEL event is determined by the closure time of the TCVs, with a faster closure time corresponding to more severe conditions. Since a conservatively small value is assumed for the closure of the turbine stop valve for the TT event, it bounds the fastest closure time for the TCVs. Therefore, the LOEL event is bounded by the TT event presented in FSAR Tier 2, Section 15.2.2.

FSAR Tier 2, Section 15.2 states that the TT event is identified as the most limiting event with respect to primary pressure. The single main steam isolation valve closure event is the most limiting event for secondary system overpressure because of the smaller available steamline volume. The radiological consequences of the TT event are bounded by the inadvertent opening of the main steam relief or safety valve described in FSAR Tier 2, Section 15.1.4.

The FSAR states that the LOCV event is a fault that can cause a TT. The consequences of the LOCV event are bounded by those of the TT event. The TT event bounds the LOCV event, because the TT event analysis does not credit the non-safety-related turbine bypass system and assumes a conservatively small turbine stop valve closure time. The turbine bypass system is isolated upon a loss of condenser vacuum to prevent over-pressurization of the condenser.

The FSAR states that the single inadvertent MSIV closure event is characterized by a decrease in heat removal by the secondary system caused by the rapid closure of one MSIV. The inadvertent closure of one MSIV isolates steam flow from its associated SG. This condition causes pressure and temperature to increase in the isolated SG, the main steamline, and consequently, in the affected RCS loop.

These events potentially challenge three acceptance criteria:

- The RCS overpressurization limit
- The secondary side overpressurization limit
- SAFDLs, specifically the DNB

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: There are no Technical Specifications for this area of review.

15.2.1.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Sections 15.2.1, “Loss of External Load,” 15.2.2, “Turbine Trip,” 15.2.3, “Loss of Condenser Vacuum,” 15.2.4, “Closure of Main Steam Isolation Valve (BWR),” and 15.2.5, “Steam Pressure Regulator Failure (Closed),” and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Sections 15.2.1-15.2.5.

1. GDC 10, as it relates to the RCS design with appropriate margin so SAFDLs are not exceeded during normal operations, including AOOs.

2. GDC 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to design of the RCS and its auxiliaries with appropriate margin so the pressure boundary is not breached during normal operations, including AOOs.

4. GDC 17, as it relates to onsite and offsite electric power systems, to ensure that safety-related SSCs function during normal operation, including AOOs. The safety function for each power system (assuming the other system is not functioning) is to provide sufficient capacity and capability so SAFDLs and RCPB design conditions are not exceeded during AOOs.

5. GDC 26, as it relates to the control of reactivity changes so SAFDLs are not exceeded during AOOs. This control is accomplished by provisions for appropriate margin for malfunctions (e.g., stuck rods).

Acceptance criteria adequate to meet the above requirements include:

1. To identify, which moderate-frequency event is the most limiting.

2. To verify whether the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure.

3. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

4. Fuel cladding integrity must be maintained by the MDNBR remaining above the 95/95 DNBR limit for PWRs and the CPR remaining above the minimum CPR safety limit for BWRs based on acceptable correlations (see FSAR Tier 2, Section 4.4) and by satisfaction of any other SAFDL applicable to the particular reactor design.

5. RG 1.105, as it relates to verifying that plant protection systems setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies.

6. RG 1.206, as it relates to verifying that event evaluations consider single failures, operator errors, and performance of non-safety-related systems.

15.2.1.4 *Technical Evaluation*

The analyses of the LOEL, TT, LOCV, and MSIVC use the previously approved non-loss-of-coolant accident analytical methodology described in AREVA TR ANP-10263P-A and the S-RELAP5 computer code to simulate the event.

The staff notes that since these events result in increasing primary system pressure prior to RT, loss of flow only after RT, and little increase in primary coolant temperature, DNBR is not challenged. The remaining evaluation focuses on primary and secondary side over-pressurization. In all cases, heat removal is achieved by heat transfer from the primary to the secondary side of the SGs, opening of redundant MSRTs to remove energy from the secondary side, and actuation of the redundant emergency feedwater to feed water to the SGs.

The staff notes that these analyses (LOEL, TT, and LOCV) are conservatively performed assuming a LOOP on TT, since LOOP causes coastdown of the RCPs, which minimizes heat transfer to the SGs and, therefore, maximizes primary system pressure. Similarly, the events are conservatively analyzed at HFP BOC, since the most positive moderator temperature coefficient maximizes the primary system heatup and, therefore pressure prior to RT. In addition to the assumed LOOP, the worst single failure is taken as failure of one MSRT to open,

since this minimizes secondary side heat removal and maximizes primary system pressure. The single main steam isolation valve closure event is the most limiting event for secondary system overpressure. In contrast to the TT, the MSIVC event loss of offsite power is not assumed since continued operation of the RCPs maximizes heat transfer to the secondary side.

Events

The TT bounds the loss of external load and the loss of condenser vacuum. These events all result in a TT and consequential RT with the TT causing the fastest system response and greatest primary system pressure increase. The staff notes that the dynamics of the event are controlled by the closure of the turbine stop valves in 0.1 sec.

The staff notes that in contrast to the TT, the main steam isolation valve closure is conservatively analyzed using conditions that maximize SG secondary side pressure. Only one valve is assumed to close due to the design of the control system. The remaining three SGs continue to remove decay heat. LOOP is not assumed, since LOOP results in RCP trip and continued operation of the RCPs maximize heat transfer to the SG secondary side and, therefore, secondary side pressure. Therefore, the staff observes that the initial conditions have been selected to maximize the severity of the analysis.

The applicant compares radiological consequences amounts to different types of events. In RAI 34, Question 15.02.01-15.02.05-1, the staff requested that the applicant justify its approach of comparing one event to another in different event types. In an October 31, 2008, response to RAI 34, Question 15.02.01-15.02.05-1, the applicant stated: The radiological consequence evaluation for an AOO addresses an event that results in the largest amount of radioactive releases to the environment and bounds any other AOO. The AOO radiological evaluation addresses the radiological impact associated with a main steam safety valve failure event. The staff's evaluation of radiological consequences are discussed at length in Section 15.0.3 of this report.

The FSAR presents analysis of the TT with LOOP. In order to better understand the system dynamics, in RAI 34, Question 15.02.01-15.02.05-2, the staff requested information about TT with offsite power available. In a September 24, 2008, response to RAI 34, Question 15.02.01-15.02.05-2, the applicant stated: Assuming LOOP coincident with the RT results in a coastdown of the RCPs during the early portion of the TT event. The reduction in RCS flow results in a decrease in the heat transfer from the RCS to the secondary system. The reduction in heat transfer to the secondary system exacerbates the heatup and pressurization of the RCS. Therefore, assumption of LOOP is conservative for the TT peak RCS pressure case. Conversely, assuming no-LOOP is conservative for the TT peak secondary pressure case. Continued operation of the RCPs maximizes the energy transfer from the RCS to the secondary side. The TT peak secondary pressure case is bounded by the results of the main steam isolation valve closure (MSIVC) event. The staff concurs in this explanation of the system dynamics and, therefore, the staff considers RAI 34, Question 15.02.01-15.02.05-2 resolved.

In RAI 34, Question 15.02.01-15.02.05-3, the staff requested that the applicant provide additional information regarding the assumed worst single failure. In FSAR Tier 2, Section 15.2.2, the analysis assumes single failure of a secondary side MSRIV. All three pressurizer safety valves are assumed to function. With respect to maximum primary pressure, the staff requested that the applicant explain why the secondary side safety valve failure to open is more restrictive than a primary side safety valve failure to open. In a September 2, 2008, response to RAI 34, Question 15.02.01-15.02.05-3, the applicant stated that the PSRVs are not

active components during hot RCS conditions and are not subject to single failure assumptions, whereas the main steam relief isolation valves are subject to single failure assumptions. FSAR Tier 2, Section 5.2.2.9, “System Reliability,” states: During hot RCS conditions, the PSRVs are considered a passive device. The spring operated pilot valves are designed in accordance with the requirements of ASME Section III, NB-7511.1. With successful operation of the pilot valve, a large differential pressure reliably opens the main relief disk, which relieves RCS pressure. A single failure is not postulated because the PSRVs do not require additional motive force to perform their design safety function during RCS hot conditions. The staff finds the response acceptable, since it complies with the ASME B&PV Code, Section III. Therefore, the staff considers RAI 34, Question 15.02.01-15.02.05-3 resolved.

In order to assess conformance with the acceptance criteria for the loss of external load, TT, and loss of condenser vacuum, in RAI 34, Question 15.02.01-15.02.05-4, the staff requested that the applicant provide the MDNBR as a function of time. In a September 24, 2008, response to RAI 34, Question 15.02.01-15.02.05-4, the applicant stated: From a DNBR perspective, the loss of external load, TT, and loss of condenser vacuum events are similar. The initiating event causes an immediate closure of the turbine control valve or turbine stop valve, which results in a sudden increase in pressure and gradual increase in core inlet temperature. The TT event is used to provide an evaluation of DNBR for these events. The maximum heat flux occurs at 0.00 seconds. Rod insertion occurs at approximately 7.4 seconds. At 0.00 seconds, the MDNBR is restricted by the DNB LCO. At 7.4 seconds, the steady mass flow (99.4 percent of initial), large increase in pressure 1,847 kPa (268 psi increase), and decreased heat flux (98.9 percent of initial) will offset the degradation due to the increased temperature 3.08 °C (5.55 °F increase). The DNBR will not degrade in comparison to the initial condition of the transient. A representative plot of the MDNBR as a function of time for the closure of a main steam isolation valve was provided in the response to RAI 34, Question 15.02.01-15.02.05-6. The assessment is acceptable, since it describes the systems response and parameters, which affect DNBR. Since the DNBR does not degrade compared to the initial condition, the staff considers RAI 34, Question 15.02.01-15.02.05-4 resolved.

The FSAR analysis of the single MSIV closure shows flow oscillations in the steamlines. In RAI 34, Question 15.02.01-15.02.05-5, the staff requested that the applicant provide an explanation of the behavior. In a September 2, 2008, response to RAI 34, Question 15.02.01-15.02.05-5, the applicant stated that the oscillations in the three unaffected steamlines are due to the fast closure of the turbine valve. The main steam isolation valve in steamline 1 is assumed to fail closed in the MSIVC analysis. The fast closure of the turbine valve results in the pressure waves reverberating in the three unaffected steamlines. The explanation describes the system response and phenomenology and is acceptable. This behavior has been observed at operating plants. Therefore, the staff considers RAI 34, Question 15.02.01-15.02.05-5 resolved.

In order to assess conformance with the acceptance criteria for the single MSIVC event, in RAI 34, Question 15.02.01-15.02.05-6, the staff requested that the applicant provide a plot of MDNBR as a function of time. In a September 24, 2008, response to RAI 34, Question 15.02.01-15.02.05-6, the applicant provided the requested plot. The plot shows little change prior to RT and increasing margin following RT. This conforms to the change of system parameters in the event. Therefore, the staff considers RAI 34, Question 15.02.01- 15.02.05-6 resolved.

In RAI 34, Question 15.02.01-15.02.05-7, the staff requested that the applicant provide a description of how DNBR calculations were done for the single MSIV closure event as addressed in FSAR Tier 2, Section 15.2.4. In a September 24, 2008, response to RAI 34, Question 15.02.01-15.02.05-7, the applicant stated: The analysis is primarily a deterministic LYNXT calculation. To provide an evaluation of the MDNBR for the affected region of the core, the affected loop's temperature is applied; the affected loop's temporal response is adjusted to account for loop transport time (from sensor to affected core region). To conservatively model the power response, the fuel rod surface heat flux is assumed constant up to the point of rod insertion due to the RT on high SG pressure. The core inlet mass flow and core exit pressure are provided from S-RELAP5. A deterministic DNBR evaluation is performed using the LYNXT code at the starting condition and point of rod insertion. The DNBR degradation calculated with LYNXT is near negligible (-0.03) from the initial condition, resulting in a conservatively calculated MDNBR of 2.08. Since the predicted minimum DNBR provides substantive margin to the acceptance criteria the staff find the response acceptable and consider RAI 34, Question 15.02.01—15.02.05-7 resolved.

The inadvertent closure of a single MSIV described in the FSAR Tier 2, Section 15.2.4.3.2, "DNBR Analysis Results," implies that a three dimensional (3-D) kinetics calculation was done. In RAI 34, Question 15.02.01-15.02.05-8, the staff requested that the applicant provide an explanation of the modeling. In a September 24, 2008, response to RAI 34, Question 15.02.01-15.02.05-8, the applicant stated that the analysis uses cold leg temperatures in combination with point kinetics and a conservatively negative MTC to determine reactivity in the S-RELAP5 model. This combination results in a slight increase in the reactor power and a more conservative analysis. The staff confirmed that FSAR Revision 1 provided a revised description. The staff reviewed the changes made to FSAR Tier 2, Section 15.2.4.3.2, "DNBR Analysis Results," and found the incorporated information acceptable. Therefore, the staff considers RAI 34, Question 15.02.01-15.02.05-8 resolved.

Anticipatory Reactor Trip on Turbine Trip

Although the analysis of the TT takes credit only for safety-related equipment and is, therefore, acceptable, the staff was concerned about the lack of an anticipatory RT on TT. In lieu of this anticipatory trip, the U.S. EPR relies on a non-safety-related partial RT for load rejection events including TT. The applicant does not take credit for this trip in the safety analyses. The staff requested that the applicant provide additional information to justify the lack of the anticipatory trip in RAI 241, Question 15.02.05-15.02.05-9. In an August 5, 2009, response to RAI 241, Question 15.02.05-15.02.05-9, the applicant stated that other than 10 CFR 50.34(f)(2)(xxiii)(Issue II.K.2.10), which applies only to Babcock and Wilcox (B&W) plants, anticipatory trips are not required by regulation. The applicant explained that although not a full RT, the U.S. EPR partial trip feature functions as an anticipatory trip to mitigate potential challenges to the plant and ESF. The partial trip results in less plant upset than a full RT. For the loss of condenser vacuum event in which the turbine bypass system is unavailable, the partial trip still prevents the actuation of the PSRVs. The response explained the function of the partial trip and confirmed that it reduces the likelihood of challenging the PSRV, and is therefore acceptable. Therefore, the staff considers the response to RAI 241, Question 15.02.05-15.02.05-9 acceptable and the issue closed.

15.2.1.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.2.1.6 *Conclusions*

The staff concludes that the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure in that the analysis demonstrates that the maximum RCS pressure of 19.203 MPa (2,785.2 psia) remains below 110 percent of the RCS design pressure of 19.327 MPa (2,803.2 psia), thereby satisfying overpressure acceptance criteria for the RCS.

The staff concludes that the analyses show that the opening setpoints and capacity of the MSSVs are adequate to limit peak secondary pressure to 10.625 MPa (1,541 psia), which is less than the acceptance criterion of 110 percent of the secondary system design pressure of 10.985 MPa (1,593.2 psia).

The staff concludes that he analyses assume single failures, operator errors, and performance of non-safety-related systems in conformance with RG 1.206.

The staff concludes that the event evaluation considers single failures in accordance with RG 1.53.

The staff concludes that RG 1.105 has been satisfied in that the plant protection systems setpoints assumed in the transients analyses are selected with allowance for measurement inaccuracies, and the plant protection systems setpoints assumed in the transients analyses were selected with allowance for measurement inaccuracies due to harsh environment effects as delineated in RG 1.105.

The staff concludes that DNBR SAFDL is not challenged.

Therefore, the staff finds that:

1. GDC 10 is met in that the safety analyses show that SAFDLs are not exceeded during normal operations, including AOOs.

2. GDC 13 is met in that the safety analyses include safety-related instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety.

3. GDC 15 is met as it relates to the analyses to demonstrate that the pressure boundary is not breached during normal operations, including AOOs.

4. GDC 17 is met as it relates to onsite and offsite electric power systems, in that the safety analyses show that with and without off site power SAFDLs and RCPB design conditions are not exceeded.

5. GDC 26 is met in that the analysis shows that reactivity control is accomplished with timely rod insertion and margin for malfunctions (e.g., stuck rods).

15.2.2 Turbine Trip

The staff's review of FSAR Tier 2, Section 15.2.2, "Turbine Trip," is documented in Section 15.2.1 of this report.

15.2.3 Loss of Condenser Vacuum

The staff's review of FSAR Tier 2, Section 15.2.3, "Loss of Condenser Vacuum," is documented in Section 15.2.1 of this report.

15.2.4 Inadvertent Main Steam Isolation Valve Closure

The staff's review of FSAR Tier 2, Section 15.2.4. "Inadvertent Main Steam Isolation Valve Closure," is documented in Section 15.2.1 of this report.

15.2.5 Steam Pressure Regulator Failure

This event is not applicable to the U.S. EPR design.

15.2.6 Loss of Non-Emergency AC Power to the Station Auxiliaries

15.2.6.1 *Introduction*

A loss of non-emergency ac power (LNEP) to the station auxiliaries is a malfunction initiated by a complete loss of either the external (offsite) grid or the onsite ac distribution system.

15.2.6.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.2.6, summarized here, in part, as follows:

The FSAR states that the short term consequence of the LNEP event is bounded by the complete loss of coolant flow event described in FSAR Tier 2, Section 15.3.2, "Complete Loss of Forced Reactor Coolant Flow," assuming LOOP or LNEP as the initiating event. The subsequent evolution of the event after RT is similar to and bounded by the loss of normal feedwater flow (LNFF) scenario with LOOP, which causes a complete loss of RCS flow; see FSAR Tier 2, Section 15.2.7 for the long-term plant response to an LNEP event.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: There are no Technical Specifications for this area of review.

15.2.6.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.2.6, "Loss of Non-emergency AC Power to the Station Auxiliaries," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.2.6.

1. GDC 10, as it relates to the RCS design with appropriate margin so SAFDLs are not exceeded during normal operations, including AOOs.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to design of the RCS and its auxiliaries with appropriate margin so the pressure boundary is not breached during normal operations, including AOOs.

4. GDC 26, as it relates to the control of reactivity changes so SAFDLs are not exceeded during AOOs. This control is accomplished by provisions for appropriate margin for malfunctions (e.g., stuck rods).

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity should be maintained by keeping the MDNBR above the 95/95 DNBR limit for PWRs based on acceptable correlations (see NUREG-0800, Section 4.4).

3. To identify, which moderate-frequency event is the most limiting.

4. To verify whether the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure.

5. RG 1.105, as it relates to verifying that plant protection systems setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies.

6. RG 1.53, as it relates to identifying and assuming in the analysis the most limiting plant systems single failure, as defined in the “Definitions and Explanations” of 10 CFR Part 50, Appendix A.

7. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

15.2.6.4 *Technical Evaluation*

Loss of non-emergency power to the station auxiliaries will cause loss of power to all the RCPs causing an early RT on low flow, as well as loss of power to the feedwater and condensate pumps causing low inventory in the SGs and a later RT. Multiple RT signals will occur. Following trip, decay heat will be removed by opening the MSRT, with SG inventory maintained by the emergency feedwater system. The analysis assumes loss of one EFW division as the most limiting single failure and is conservatively performed with BOC reactivity coefficients and conservative scram reactivity.

The analysis of the LNEP uses the approved non-loss-of-coolant accident analytical methodology described in AREVA TR ANP-10263P-A, “Codes and Methods Applicability Report of the U.S. EPR,” August 2007, which utilizes the S-RELAP5 computer code to simulate the U.S. EPR reactor system-level response. Input values were conservatively chosen.

In RAI 34, Question 15.02.06-1, the staff noted that insufficient information was provided to assess the conclusions in the FSAR. The staff requested that the applicant provide a sequence of events explaining the functioning of the RCPs, feedwater pumps, and other auxiliaries, which are lost upon loss of non-emergency power and explaining why this event is bounded. In a September 24, 2008, response to RAI 34, Question 15.02.06-1, the applicant stated that the loss of non-emergency power, described in the FSAR, is initiated by either a complete LOOP coincident with a TT, or a loss of onsite power. Diesel generators are started and provide electric power to vital loads. The sensible and decay heat loads are handled by the steam relief and emergency feedwater systems. Different segments of this event have similarities to the complete loss of RCS flow events and complete loss of normal feedwater flow.

The staff reasoned that at event initiation, power to the RCPs, and main feedwater and condensate pumps is lost. The time constant of the SGs is considerably larger than the time constant of the fuel and RCS. Therefore, the early part of the transient is bounded by the loss of flow event, and the metric is DNBR. After trip, the event is bounded by the loss of feedwater with LOOP, and the metric is primary and secondary side pressure. The relative timing is illustrated by FSAR Tier 2, Table 15.3-4, "Complete Loss of Forced Reactor Coolant Flow Event-Sequence of Events," which shows RT on low RCP speed at 1.4 sec; while in contrast, FSAR Tier 2, Table 15.2-9, "LNFF-Sequence of Events" shows the SG water level reaches low narrow range (NR) level RT at 42.1 sec. Therefore, the staff concluded that the LNEP is bounded by the loss of flow event and the loss of feedwater event and that the applicant's response adequately reflects the system response and is acceptable. Therefore, the staff considers RAI 34, Question 15.02.06-1 resolved.

15.2.6.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.2.6.6 *Conclusions*

The staff concludes that the bounding analyses show that SAFDLs are not violated and that primary and secondary pressures are maintained at less than 110 percent of design limits. The staff concludes that the transient is bounded by the loss of forced flow, which results in the MDNBR and the loss of feedwater, which results in the maximum primary and secondary system pressures. LNEP numerical analyses are not presented in the FSAR. The staff concludes that the bounding analyses provided by the applicant assume single failures, operator errors, and performance of non-safety-related systems in conformance with RG 1.206.

The staff concludes that the bounding analyses consider single failures in conformance with RG 1.53.

The staff finds that conformance with RG 1.105 has been satisfied in that the bounding analyses consider protection systems setpoints with allowance for measurement inaccuracies.

Therefore, the staff finds:

1. GDC 10 is met in that the safety analyses show that SAFDLs are not exceeded during normal operations, including AOOs.

2. GDC 13 is met in that the safety analyses include safety-related instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety.

3. GDC 15 is met as it relates to the analyses to demonstrate that the pressure boundary is not breached during normal operations, including AOOs.

4. GDC 26 is met in that the analysis shows that reactivity control is accomplished with timely rod insertion and margin for malfunctions (e.g., stuck rods).

15.2.7 Loss of Normal Feedwater Flow

15.2.7.1 *Introduction*

The loss of normal feedwater is an AOO resulting in a postulated complete termination of main feedwater flow. This condition could be caused by a LOOP or a malfunction in the MFW control system or equipment.

15.2.7.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.2.7, summarized here, in part, as follows:

The sudden loss of MFW flow while the plant continues to operate at power results in a reduction in SG heat removal and an increase in reactor coolant temperature and pressure. RT is initiated on a low SG level signal. If available, the non-safety-related turbine bypass system opens to control secondary side pressure. If not available, SG pressure increases until the safety-related MSRTs open. At the low SG level emergency feedwater setpoint, EFW and isolation of the SG blowdown line is initiated. With the delivery of EFW, the plant is stabilized.

The applicant's FSAR analysis concludes that the regulatory acceptance criteria are met, and there are no radiological consequences or releases from this event. The LNFF event is bounded by the inadvertent opening of a main steam safety or relief valve event (FSAR Tier 2, Section 15.1.4) for radiological consequences.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: There are no Technical Specifications for this area of review.

15.2.7.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Sections 15.2.7, "Loss of Normal Feedwater Flow," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.2.7.

1. GDC 10, as it relates to the RCS design with appropriate margin so SAFDLs are not exceeded during normal operations, including AOOs.

2. GDC 13, as to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to design of the RCS and its auxiliaries with appropriate margin so the pressure boundary is not breached during normal operations, including AOOs.

4. GDC 17, as it relates to onsite and offsite electric power systems, to ensure that safety-related SSCs function during normal operation, including AOOs. The safety function for each power system (assuming the other system is not functioning) is to provide sufficient capacity and capability so SAFDLs and RCPB design conditions are not exceeded during AOOs.

5. GDC 26, as it relates to the control of reactivity changes so SAFDLs are not exceeded during AOOs. This control is accomplished by provisions for appropriate margin for malfunctions (e.g., stuck rods).

6. 10 CFR 50.34(f)(1)(ii)1 and 10 CFR 50.34(f)(2)(xii), as they relate to the performance requirements of the auxiliary feedwater system for the loss of normal feedwater flow event.

Acceptance criteria adequate to meet the above requirements include:

1. Identification of which moderate-frequency event is the most limiting.

2. Verification of whether the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure.

3. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

4. Fuel cladding integrity shall be maintained by ensuring that the MDNBR remains above the 95/95 DNBR limit for PWRs, and the critical power ratio (CPR) remains above the maximum critical power ration (MCPR) safety limit for BWRs based on acceptable correlations (see FSAR Tier 2, Section 4.4), as well as by satisfaction of any other SAFDL that may be applicable to the particular reactor design. There is sufficient capacity for long term decay heat removal for the plant to reach a stabilized condition.

5. RG 1.105, as it relates to verifying that plant protection systems setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies.

6. RG 1.206, as it relates to verifying whether the event evaluation considers single failures, operator errors, and performance of non-safety-related systems.

7. TMI Action Plan Item II.K.2.19 of NUREG-0737, “Sequential Auxiliary Feedwater Flow Analysis.”

15.2.7.4 *Technical Evaluation*

The analysis of the LNFF event uses the approved non-loss-of-coolant accident analytical methodology described in AREVA TR ANP-10263P-A and the S-RELAP5 computer code to simulate the U.S. EPR reactor system-level response. The staff confirmed that the analysis was performed with conservative input values and assumptions as discussed below.

FSAR Tier 2, Section 15.2.7 states that SAFDLs are not challenged by this event. The fuel melt SAFDL is not challenged, because the reactor power never increases beyond its initial value. The DNB SAFDL is not challenged, because the subcooling margin, defined as the difference between the actual temperature and the temperature to boil, remains greater than 13.9 °C (25 °F) throughout the event. The staff concurs with the assessment provided by the applicant in FSAR Tier 2, Section 15.2.7.

FSAR Tier 2, Section 15.2.7 states that the worst single failure is failure of one EFW division. The staff concurs since the loss of normal feedwater is compounded by loss of an EFW train. The applicant's parametric studies show that: 1) Conditions at HFP BOC, minimize SG decay heat removal (i.e., minimize SG inventory); while 2) HFP EOC conditions with pressure control (operating presssurizer sprays) and LOOP maximize pressurizer overfill; and 3) HFP BOC conditions, without pressure control (no pressurizer spray) and with LOOP maximizes RCS pressure.

FSAR Tier 2, Section 15.2.7 states that the peak RCS pressure for the loss of normal feed (LONF) event is bounded by the TT event. The staff concurs, since for the LONF event, TT does not occur until after RT; whereas, in the TT event, TT occurs 7 seconds before RT. Therefore, energy addition to the RCS is greater for the TT event, implying a higher peak RCS pressure for the TT event.

The peak secondary pressure for the LONF event is bounded by the MSIVC event. In the LONF event, a significant amount of liquid is removed from the secondary side prior to TT; whereas, in the MSIVC event, steam not liquid is removed prior to isolation of the affected SG. Thus, the vapor volume available to be compressed after TT is larger for the LONF event than it is for the MSIVC event, implying a lower peak secondary pressure for the LONF event.

The applicant presented an analysis in FSAR Tier 2, Section 15.2.7, showing that the EFW system provides adequate decay heat removal for the case with EFW being delivered to only two SGs and the RCPs running.

The applicant committed to doing SG nodalization sensitivity studies for this event in a March 28, 2007, response to RAI 8 on TR ANP-10263P-A. The results of these sensitivity studies are presented in Small Break LOCA and Non-LOCA Sensitivity Studies and Methodology. The number of fluid cells representing the SG tubes and the SG boiler region were doubled for the sensitivity study. The results for the base LONF simulation and the LONF simulation with more SG nodes were compared. The comparison showed that the two simulations produced similar results both before and after RT. The time of RT when the number of SG nodes was increased was earlier, because the high SG level trip was reached sooner. Based on the comparison, the applicant concluded that the base SG noding is acceptable for LONF simulations. The staff concurs, because the base case nodalization produces slightly more conservative results.

In order to assess conformance with the DNBR acceptance criteria for this event, in RAI 34, Question 15.02.07-1, the staff requested that the applicant provide a figure showing MDNBR as a function of time for this event. In a September 24, 2008, response to RAI 34, Question 15.02.07-1, the applicant provided the requested figure, which shows only a slight degradation in DNBR margin from the steady state operational margin, and the MDNBR remains above DNBR limits set forth in FSAR Tier 2, Section 4.4. The applicant's response is acceptable. Therefore, the staff considers RAI 34, Question 15.02.07-1 resolved.

15.2.7.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.2.7.6 *Conclusions*

The staff concludes that the applicant performed the LONF analysis using approved methods. The most severe single failure is the loss of an EFW train. The staff notes that another EFW train was assumed to be out for maintenance. The staff agrees that SAFDLs are not challenged by this event and that the peak primary and secondary pressures for this event are within acceptable limits. Specifically, FSAR Tier 2, Figure 15.2-29, "Loss of Normal Feedwater-PZR Pressure," and FSAR Tier 2, Figure 15.2-39, "Loss of Normal Feedwater- SG Pressure," show that the primary system pressure remains below the limit of 19.3 MPa (2,803 psia), and the secondary pressure remains below the limit of 11 MPa (1,593 psia). The staff find that the applicant has also demonstrated that the EFW system provides adequate long term decay heat removal. Therefore, the staff finds that the SRP acceptance criteria for the LONF event are met.

The staff concludes that the event evaluation considers single failures in conformance with RG 1.53, and performance of non-safety-related systems conforms to the guidance contained in RG 1.206.

Therefore, the staff finds that:

1. GDC 10, as it relates to the RCS design is met in that SAFDLs are not challenged.

2. GDC 13 is met in that the safety analysis takes credit for only safety-related instrumentation.

3. GDC 15, as it relates to design of the RCS is met in that the analysis shows that primary system pressure is maintained at less than 110 percent of design.

4. GDC 17 is met in that the analysis has been preformed with and without onsite and offsite electric power systems, and shows that acceptance criteria are met.

5. GDC 26, as it relates to the control of reactivity changes is met in that the analysis has been performed with conservative values of rod worth and rod insertion time and the most reactive rod has been assumed stuck in the fully withdrawn position.

6. 10 CFR 50.34(f)(1)(ii)1 and 10 CFR 50.34(f)(2)(xii)) are met in that an automatic safety-related emergency feedwater system is provided.

15.2.8 Feedwater Line Breaks Inside and Outside Containment

15.2.8.1 *Introduction*

A feedwater line break (FWLB) is a postulated accident resulting from a rupture in a feedwater line large enough to prevent the addition of sufficient feedwater to the SGs to maintain secondary side fluid inventory in the SGs.

15.2.8.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.2.8, summarized here, in part, as follows:

FSAR Tier 2, Section 15.2.8 states that a break upstream of the main feedwater line check valve affects the plant only as a loss of main feedwater. This scenario is bounded by the event evaluations for the loss of normal feedwater (FSAR Tier 2, Section 15.2.7) and loss of non-emergency ac power (FSAR Tier 2, Section 15.2.6). Considering breaks downstream of the FW check valves, and depending on the size and location of the break and the plant operating conditions at the time of the break, the break could cause either a RCS cooldown due to excessive energy discharge through the break or a RCS heat up. The FWLB scenarios that cause a cooldown of the RCS are bounded by the analyses described in FSAR Tier 2, Section 15.1.5 for the MSLB event.

Results are presented for three cases: First representative small feedwater line breaks; second a case which maximizes RCS pressure; and third a case which maximizes main steam system pressure.

The applicant's FSAR analysis concludes that the regulatory acceptance criteria are met, and there are no radiological consequences or releases from this event. The FWLB event is bounded by the MSLB event (FSAR Tier 2, Section 15.1.5) with regard to SAFDLs. The radiological consequences of this event are described in FSAR Tier 2, Section 15.0.3.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 15.2.8 are given in FSAR Tier 2, Chapter 16, Section 3.4.1.

15.2.8.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.2.8, "Feedwater System Pipe Break Inside and Outside Containment (PWR)," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.2.8.

1. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

2. GDC 17, as it relates to onsite and offsite electric power systems, to ensure that safety-related SSCs function during normal operation, including AOOs. The safety function for each power system (assuming the other system is not functioning) is to provide sufficient capacity and capability so SAFDLs and RCPB design conditions are not exceeded during AOOs.

3. GDC 27 and GDC 28, as they relate to the RCS design with appropriate margin so acceptable fuel design limits are not exceeded and core cooling capability is maintained.

4. GDC 31, as it relates to RCS design with sufficient margin so the boundary is nonbrittle and the probability of fracture propagation is minimized.

5. GDC 35, as it relates to design of the RCS and its auxiliaries for abundant emergency core cooling.

6. 10 CFR Part 100, as it relates to calculated doses at the site boundary.

7. Requirements for maintenance of adequate decay heat removal by the AFWS are in 10 CFR 50.34(f)(1)(ii), (TMI issue II E 1.1) and 10 CFR 50.34(f)(2)(xii), (TMI issue II E 1.2).

8. Requirements for RCP operation are in 10 CFR 50.34(f)(1)(iii), (TMI issue 2 K 2).

Acceptance criteria adequate to meet the above requirements include:

1. Identification of which moderate-frequency event is the most limiting.

2. Verification of whether the predicted plant response for the most limiting event satisfies the specific criteria for fuel damage and system pressure.

3. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design pressures (ASME B&PV Code, Section III) for low-probability events and below 120 percent for very low-probability events like double-ended guillotine breaks.

4. The potential for core damage is evaluated for an acceptable MDNBR remaining above the 95/95 DNBR limit for PWRs based on acceptable correlations (see NUREG-0800, Section 4.4). If the DNBR falls below these values, fuel failure (rod perforation) must be assumed for all rods not meeting these criteria unless, from an acceptable fuel damage model (see NUREG-0800, Section 4.2) including the potential adverse effects of hydraulic instabilities, fewer failures can be shown to occur. Any fuel damage calculated to occur must be of sufficiently limited extent that the core remains in place and intact with no loss of core cooling capability.

5. RG 1.105, as it relates to verifying that plant protection systems setpoints assumed in the transients analyses are selected with adequate allowance for measurement inaccuracies.

6. RG 1.206, as it relates to verifying that event evaluations consider single failures, operator errors, and performance of non-safety-related systems.

15.2.8.4 *Technical Evaluation*

The analysis of the FWLB event scenarios that result in a heat up of the RCS uses the approved non-loss-of-coolant accident analytical methodology described in AREVA TR ANP-10263P-A and the S-RELAP5 computer code to simulate the event.

The applicant has investigated a spectrum of break sizes ranging from 1.7 cm^2 (0.0184 ft^2) to 857 cm^2 (0.922 ft^2). The minimum area was chosen to be the largest break area that can exist and yet have the MFW system supply full flow to a SG. The maximum break is the area of the

inlet to the MFW half-ring header in a SG. The applicant has considered cases with and without LOOP on RT. Three cases are presented in the FSAR. The first case, 1.7 cm^2 (0.0184 ft^2) break, is the case that gives the highest pressurizer level; the second case, 386 cm^2 (0.415 ft^2) break, is the one which gives the highest RCS pressure; and the third case, 857 cm^2 (0.922 ft^2) break, is the one which gives the highest secondary pressure. The applicant has assumed complete loss of MFW at the start of each simulation. The staff finds that this is a conservative assumption, because even for the largest break some MFW could be supplied to the unaffected SGs if the MFW pumps continued to operate. Additional conservatism was incorporated into all simulations by assuming degraded environment instrument uncertainties to all setpoints.

For the pressurizer overfill and RCS peak pressure cases, the FSAR analysis takes the worst single failure is the loss of an EFW division. Another division is assumed to be out for maintenance. Furthermore, one of the active divisions is assumed to be connected to the affected SG and all of its flow is assumed to be lost out the break. The staff concurs that this is the worst failure since loss of normal feedwater to the steam generators is compounded by the loss of emergency feedwater. Thirty minutes after RT, the operator is assumed to align this EFW division so it provides flow to an unaffected SG in which EFW is not operating. The staff notes that credit for operator action after 30 minutes is consistent with human action guidance.

FSAR Tier 2, Section 15.2.8 shows that for the secondary peak pressure case, the worst failure is the inoperability of an MSRT connected to an unaffected SG. The staff concurs since failure to remove energy increases pressure. For this case, there are three EFW divisions available (one is out for maintenance), but one of those is assumed to be connected to the affected SG and all its flow is lost to the break. As with the other simulations, it is assumed that 30 minutes after RT the operator redirects this EFW to an unaffected SG not already receiving EFW.

The applicant simulated the first 7,000 seconds of the pressurizer overfills case using approved methods. The simulation showed that the plant reached a stable cooldown after 2,000 seconds.

The RCS peak pressure simulation also showed that the plant reached a stable cooldown after 2,000 seconds.

The staff notes that sub-cooling margin for this case is always greater than the margin during normal operation, and reactor power is never greater than the initial value. Therefore, SAFDLs are not challenged. The peak pressure is calculated to be 18.450 MPa (2,676 psia), which is below the 110 percent design value of 19.327 MPa (2,803 psia).

The secondary peak pressure simulation shows that the plant is in a stable mode by 300 seconds with all decay heat being removed via steam release through the MSRTs and the isolated SG's relief valves. The maximum secondary side pressure is 10.555 MPa (1,531 psia), which is below the 110 percent design value of 10.985 MPa (1,593 psia). Since the sub-cooling margin is always greater than the normal operation margin, and reactor power is never greater than the initial value of 4,612 MWt, so SAFDLs are not challenged by this event.

The feedwater line break analyses presented by the applicant assumed operator action 30 minutes after RT, but according to Section 5.2.2.2.3.1 of AREVA TR ANP-10263P-A, operator action is not credited until 1 hour after RT. The assumption of operator action after 30 minutes has been made for all other Chapter 15 events in which operator action was assumed. The assumption of longer time for operator action is conservative. The staff determines that the assumption is also acceptable for this event (Reference NUREG-0700).

Fourteen RAIs were issued during the review of the FWLB event. All of these RAIs were issued to obtain clarification of how the FWLB analysis was conducted or to address editorial issues. The staff reviewed the applicant's responses for all of the RAIs and determined that the applicant's responses adequately addressed the staff's concerns. Therefore, all of the applicant's responses are acceptable. The staff's RAIs and the applicant's responses are given below. Figures provided in the applicant's responses have been omitted.

In RAI 34, Question 15.02.08-1, the staff requested that the applicant provide a description of the modeling methodology used to analyze the reactor behavior during feedwater pipe break transients. In a September 2, 2008, response to RAI 34, Question 15.02.08-1, the applicant stated that AREVA TR ANP-10263P-A and EMF-2310P-A, Revision 1, "SRP Chapter 15 Non-LOCA Methodology for Pressurized Water Reactors," describe the applicability and general methodology used for the feedwater line break analysis. The staff finds use of previously approved methods acceptable and, therefore, considers RAI 34, Question 15.02.08-1 resolved.

In order to assess conformance with acceptance criteria, in RAI 34, Question 15.02.08-2, the staff requested that the applicant provide the calculated DNBR and amount of fuel failures for the FWLB. In a September 24, 2008, response to RAI 34, Question 15.02.08-2, the applicant stated that the FSAR Tier 2, Section 15.2.8.1, "Identification of Causes and Event Description," states that the feedwater line break scenarios that cause a cooldown of the RCS are bounded by the analysis described in FSAR Tier 2, Section 15.1.5, "Steam Line Break." During the post-scram peak return to power with a stuck control rod for the FSAR Tier 2, Section 15.1.5 event, 1.24 percent of the fuel assemblies are predicted to fail by exceeding the clad strain limit. No fuel failures result from fuel melt. The applicant provided the requested plot, which shows margin to DNBR. This transient event is not DNB or LPD limiting and does not cause an in-core trip. The applicant's response has provided the requested information, and is therefore acceptable.

In RAI 34, Question 15.02.08-3, the staff requested that the applicant clarify the labeling of several figures. In a September 24, 2008, response to RAI 34, Question 15.02.08-3, the applicant stated that the figures were incorrect and would be replaced with corrected figures as shown in the response to RAI 34, Question 15.02.08-3. The staff confirmed that the changes were incorporated in Revision 1 of the FSAR and therefore consider RAI 34, Question 15.02.08-3 resolved.

In RAI 34, Question 15.02.08-4, the staff requested that the applicant provide the decay heat curve used in the analysis. In an October 31, 2008, response to RAI 34, Question 15.02.08-4, the applicant referred to its response to RAI 34, Question 15-7 (see Section 15.0.1 of this report) which provided the requested decay heat curve. Since the requested information was provided by the applicant in the response to RAI 34, Question 15-7, the staff considers RAI 34, Question 15.02.08-4 resolved.

In RAI 34, Question 15.02.08-5, the staff requested that the applicant clarify the predicted dynamical behavior of the system as shown in FSAR Tier 2, Figure 15.2-50, "FWLB Representative Small Break - Pressurizer Pressure." In a September 24, 2008, response to RAI 34, Question 15.02.08-5, the applicant indicated that the figure was incorrect and would be replaced with a corrected figure. The staff reviewed the corrected figure and found it to be acceptable. Similarly, in RAI 34, Question 15.02.08-6, the staff requested that the applicant clarify the predicted system transient pressure shown in FSAR Tier 2, Figure 15.2-66, "FWLB Representative Small Break - Steam Generator Maximum Pressure." In an October 31, 2008,

response to RAI 34, Question 15.02.08-6, the applicant indicated that this figure was also incorrect and would be replaced with a corrected figure. The staff reviewed the corrected figure and found it to be acceptable. In RAI 34, Question 15.02.08-7, the staff requested that the applicant clarify the values shown in FSAR Tier 2, Figure 15.2-67, "FWLB Representative Small Break − Reactivities." In a September 24, 2008, response to RAI 34, Question 15.02.08-7, the applicant stated that the figure was incorrect and would be replaced with a corrected figure. The staff reviewed the corrected figure and found it to be acceptable. In RAI 34, Question 15.02.08-8, the staff requested that the applicant clarify the values shown in FSAR Tier 2, Figure 15.2-68, "FWLB Representative Small Break - Liquid Volume Fraction in Pressurizer Dome." In a September 24, 2008, response to RAI 34, Question 15.02.08-8, the applicant stated that the figure was incorrect and would be replaced with a corrected figure. The staff reviewed the corrected figure and found it to be acceptable. In RAI 34, Question 15.02.08-10, the staff requested that the applicant clarify the values shown in FSAR Tier 2, Figure 15.2-69, "FWLB Maximum RCS Pressure Case − Reactor and Total Steam Generator Power." In a September 24, 2008, response to RAI 34, Question 15.02.08-10, the applicant stated that the figure was incorrect and would be replaced with a corrected figure. The staff reviewed the corrected figure and found it to be acceptable. The staff confirmed that the above corrections proposed by the applicant in the responses to RAI 34, Questions 15.02.08-5, 15.02.08-6, 15.02.08-7, 15.02.08-8, and 15.02.08-10 mentioned above were contained in FSAR Revision 1. Therefore, the staff considers RAI 34, Questions 15.02.08-5, 15.02.08-6, 15.02.08-7, 15.02.08-8, and 15.02.08-10 resolved since the revised figures show the dynamical behavior of the primary and secondary systems during a feedwater line break.

To better understand the accident progression, the staff requested information about the break sizes, mass flow rates, and other parameters in the FWLB-small break, FWLB-maximum pressure case, and FWLB –secondary pressure case. In RAI 34, Question 15.02.08-9, the staff requested that the applicant provide an explanation of the criterion used for determining the worst break size for the FWLB-small break. The staff requested that the applicant provide the flow quality as well as the fluid-phase temperatures at the break opening. In addition, the staff requested that the applicant provide the liquid- and vapor-phase mass flow rate at the break and specify the break backpressure used in the calculations. In a September 2, 2008, response to RAI 34, Question 15.02.08-9, the applicant stated that a spectrum of break sizes is investigated that ranges from the minimum break size that could be overfed by the main feedwater, 17 cm^2 (0.0184 ft^2), to the area of the MFW distribution half-ring inlet, 857 cm^2 (0.922 ft^2). The representative small break results presented produce the maximum pressurizer level. The break flow is determined by S-RELAP5 utilizing the single velocity Moody critical break flow correlation. The containment pressure (i.e., break backpressure) is fixed at 101.4 kPa (14.7 psia). The applicant also provided the void fraction at the break opening and the vapor phase temperature and saturation temperature in the break upstream volume, as well as the liquid and vapor mass flows. The staff finds the applicant's response acceptable since the break flow model is conservative and, therefore, considers RAI 34, Question 15.02.08-9 resolved.

In RAI 34, Question 15.02.08-11, the staff requested that the applicant provide additional information about the FWLB maximum pressure case, similar to the information requested for the FWLB-small break case. The staff requested that the applicant provide an explanation of the criterion used for determining the worst break size for the FWLB-maximum RCS pressure case and the related calculated parameters. In a September 2, 2008, response to RAI 34, Question 15.02.08-11, the applicant stated that a spectrum of break sizes was investigated that ranged from the minimum break size that could be overfed by the main feedwater, 17 cm^2

(0.0184 ft^2), to the area of the MFW distribution half-ring inlet, 857 cm^2 (0.922 ft^2). The results of the break size cases were reviewed to determine which break size produced the maximum RCS pressure. The limiting RCS pressure case is the 385.55 cm^2 (0.415 ft^2) break (45 percent of the largest break size investigated). The applicant provided void fraction at the break opening, vapor phase temperature and saturation temperature in the break upstream volume, and liquid and vapor mass flows. The staff notes that the break flow is calculated conservatively by S-RELAP5 utilizing the single velocity Moody critical break flow correlation. The containment pressure (i.e., break backpressure) is fixed at 101 KPa (14.7 psia). The worst break is identified by analyzing a spectrum of break sizes. Therefore, the staff concludes that the analysis accurately identifies the worst break and the analysis is conservative. The staff considers RAI 34, Question 15.02.08-11 resolved.

In RAI 34, Question 15.02.08-12, the staff requested that the applicant provide the valve flow areas used to compute the mass flow rates shown in FSAR Tier 2, Figure 15.2-78, "FWLB Maximum RCS Pressure Case – Main Steam Relief Loops 1 and 2," and in FSAR Tier 2, Figure 15.2-79, "FWLB Maximum RCS Pressure Case – Main Steam Relief Loops 3 and 4." In a September 2, 2008, response to RAI 34, Question 15.02.08-12, the applicant stated that the normalized main steam relief control valve flow areas are provided in Figures 15.02.08-12-1 and 15.02.08-12-2 of the RAI response. The modeled valve full-open area for the normalization is 238.6 cm^2 (0.2598 ft^2). The main steam relief isolation valve must also be opened for flow to occur in the MSRT. The MSRCV power-dependent position has a minimum value of 40 percent open from 20 percent thermal power down to 0 percent power. When the MSRIV is open, the MSRCV position is automatically adjusted to control SG pressure. The response provides the requested information, and is therefore acceptable. The staff considers RAI 34, Question 15.02.08-12 resolved.

In RAI 34, Question 15.02.08-13, the staff requested that the applicant provide the valve flow areas used to compute the mass flow rates shown in Figure 15.2-96, "FWLB Maximum Secondary Pressure Case – Main Steam Relief Loops 1 and 2," and in Figure 15.2-97, "FWLB Maximum Secondary Pressure Case - Main Steam Relief Loops 3 and 4." In a September 2, 2008, response to RAI 34, Question 15.02.08-13, the applicant stated that the normalized main steam relief control valve flow areas are provided in Figures 15.02.08-13-1 and 15.02.08-13-2 of the RAI response. The modeled valve full-open area for the normalization is 238.6 cm^2 (0.2598 ft^2). The main steam isolation valve must also be opened for flow to occur in the MSRT. MSRIV-3[1] (for SG 3) is assumed to fail to open[1] for this case, so the MSRCV-3 position has no effect on the results. The staff reviewed the applicant's response and found that it provided the requested information. Therefore, the staff considers RAI 34, Question 15.02.08-13 resolved.

To better understand the dynamical response of the reactor, in RAI 34, Question 15.02.08-14, the staff requested that the applicant provide the reactivity components used to compute the total reactivity. In a September 24, 2008, response to RAI 34, Question 15.02.08-14, the applicant provided plots of the reactivity components and total reactivity for the Turbine Trip-Peak RCS Pressure Case and for the MSIVC Peak Secondary Pressure Case. The staff reviewed the applicant's response and found that it provided the requested information. Therefore, the staff considers RAI 34, Question 15.02.08-14 resolved.

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[1] The staff notes that MSRIV-3 is assumed to fail closed, not open, for the analysis being discussed.

15.2.8.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.2.8.6 *Conclusions*

The staff concludes that the applicant has used NRC-approved methods and conservative assumptions to analyze the FWLB event. The staff finds that the results of the analysis conform with the guidance contained in NUREG-0800, Section 15.2.8.

The staff concludes that pressure in the RCS and the main steam systems is maintained below 110 percent of design pressures for all cases, and that the RCS peak pressure of 18.4 MPa (2,676 psia) is below the 110 percent design value of 19.3 MPa (2,803 psia). The staff concludes that the main steam system peak pressure of 10.55 MPa (1,531 psia) is below the 100 percent design value of 11 MPa (1,593 psia).

The staff concludes that the analysis shows that the EFW system maintains adequate decay heat removal.

The staff concludes that SAFDLs are not challenged by small FWLB transients, the large FWLB event's response is similar to that of a small steamline break, and the analyses to verify compliance with SAFDLS for steamline break events bound the FWLB event.

The staff concludes that the protection system setpoints conform to RG 1.105 in that decalibration of instruments due to harsh environment have been taken, and that the event evaluation which considers single failures, operator errors, and performance of non-safety-related systems conforms to the guidance contained in RG 1.206.

Therefore, the staff finds that:

1. GDC 13 is met in that a safety-related instrument with appropriate range is used in the analysis.

2. GDC 17, as it relates to onsite and offsite electric power systems, in that the analysis considers sequences with and without LOOP in order to identify the most limiting sequence and confirm that criteria are met.

3. GDC 27 and GDC 28 are met in that the analysis shows that acceptable fuel design limits are not exceeded and core cooling capability is maintained.

4. GDC 31, as it relates to RCS design with sufficient margin so the boundary is nonbrittle and the probability of fracture propagation is minimized in that the analysis does not show a rapid vessel cooldown.

5. GDC 35, as it relates to design of the RCS and its auxiliaries in that the analysis shows that with one train of engineered safety features failed and another train out of service for maintenance core cooling and heat removal is provided.

6. 10 CFR Part 100, as it relates to calculated doses at the site boundary is met in that calculated doses are a small fraction of the 10 CFR Part 100 guidelines (refer to FSAR Tier 2, Section 15.0.3).

7. Requirements for maintenance of adequate decay heat removal by the AFWS are in 10 CFR 50.34(f)(1)(ii), (TMI issue II E 1.1) and 10 CFR 50.34(f)(2)(xii), (TMI issue II E 1.2) are provided by an automatic safety- related EFW).

8. Requirements for RCP operation are in 10 CFR 50.34(f)(1)(iii), (TMI issue 2 K 2) and are maintained in that RCP seal injection is maintained.

15.3 Decrease in Reactor Coolant System Flow Rate

Both AOO and postulated accidents can result from a decrease in RCS flow. These events are described in the following FSAR Tier 2, Section 15.3, "Decrease in Reactor Coolant System Flow Rate," subsections:

- FSAR Tier 2, Section 15.3.1, "Partial Loss of Forced Reactor Coolant Flow" (AOO)
- FSAR Tier 2, Section 15.3.2, "Complete Loss of Forced Reactor Coolant Flow" (AOO)
- FSAR Tier 2, Section 15.3.3, "Reactor Coolant Pump Rotor Seizure"
- FSAR Tier 2, Section 15.3.4, "Reactor Coolant Pump Shaft Break"

15.3.1 Partial Loss of Forced Reactor Coolant Flow and Complete Loss of Forced Reactor Coolant Flow

This section documents the staff's review of FSAR Tier 2, Sections 15.3.1 and 15.3.2.

15.3.1.1 *Introduction*

A partial loss of forced reactor coolant flow is an AOO that may be caused by a mechanical or electrical failure in a RCS pump motor or its power supply. A complete loss of forced reactor coolant flow is an AOO occurrence resulting in the simultaneous fault of the RCPs that may be caused by an electrical power system fault. In either event, the resulting decrease in reactor coolant flow, while the reactor is at power, degrades the core heat transfer and reduces the DNB margin. These events have similar system behavior, are addressed in the same SRP, and have the same acceptance criteria. Therefore, they are discussed together.

15.3.1.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.3.1, summarized here, in part, as follows:

The FSAR states that upon a partial loss of RCS flow, the reactor is tripped when the flow in a single loop reaches the low-low flow protection system setpoint.

Upon a complete loss of RCS flow, the reactor is tripped when the flow in multiple loops reaches the low flow or low RCP speed setpoints. The loss of forced flow in multiple RCS loops also generates a RT signal at the low flow or low RCP speed setpoints.

The applicant's FSAR analysis concludes that the regulatory acceptance criteria are met, and there are no radiological consequences or releases from this event.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 15.3.1 are given in FSAR Tier 2, Chapter 16, Section 3.4.1 and Sections 3.4.4, "RCS Loops - MODES 1 and 2," through 3.4.7, "RCS Loops - MODE 5, Loops Filled."

15.3.1.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Sections 15.3.1-15.3.2, "Loss of Forced Reactor Coolant Flow Including Trip of Pump Motor and Flow Controller Malfunctions," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Sections 15.3.1-15.3.2.

1. GDC 10 and GDC 20, as they relate to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations including AOOs.

4. GDC 17, as it relates to onsite and offsite electric power systems, to ensure that safety-related SSCs function during normal operation, including AOOs. The safety function for each power system (assuming the other system is not functioning) is to provide sufficient capacity and capability so SAFDLs and RCPB design conditions are not exceeded during AOOs.

5. GDC 20, as it relates to the reactor protection system being designed to initiate automatically the operation of appropriate systems, including the reactivity control systems, to ensure that SAFDLs are not exceeded during any condition of normal operation, including AOOs.

6. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded, including AOOs. This is accomplished by ensuring that appropriate margin for malfunctions such as stuck rods are accounted for.

Acceptance criteria adequate to meet the above requirements include:

1. To identify which moderate-frequency initiating event is the most limiting.

2. To verify that, for the most limiting initiating events, the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.

3. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

4. Fuel-cladding integrity must be maintained by the MDNBR remaining above the 95 percent probability/95 percent confidence DNBR limit for PWRs based on acceptable correlations (see NUREG-0800, Section 4.4).

5. RG 1.105, to meet the requirements of GDC 10, GDC 13, GDC 15, GDC 20, and GDC 26 with regard to their impact on the plant response to the type of transient addressed in NUREG-0800.

6. RG 1.53, as it relates to identifying and assuming in the analysis the most limiting plant systems single failure, as defined in the "Definitions and Explanations" of 10 CFR Part 50, Appendix A.

15.3.1.4 *Technical Evaluation*

The staff confirmed that the analysis uses the approved non-loss-of-coolant accident analytical methodology described in TR ANP 10263P-A and the S-RELAP5 computer code to simulate the event. The change in DNB ratio is then calculated with the LYNXT core thermal-hydraulic computer code using the RCS response from S-RELAP5 as a boundary condition.

FSAR Tier 2, Section 15.3 takes the partial loss of flow is taken to be the loss of power to a single RCP, resulting in low flow in a RCS loop. The staff confirmed that the safety-related low-low flow signal trips the reactor essentially terminating the event. The RT will result in a TT and the assumed loss of power to the remaining RCPs. The MDNBR occurs prior to the TT and tripping of the remaining RCPs so the staff notes that the assumed LOOP has little effect on this transient MDNBR. The loss of primary flow decreases heat transfer to the SGs and maximizes primary system pressure during the transient. FSAR Tier 2, Section 15.3 states that a postulated two pump coastdown is bounded by the single pump coastdown, because the two pump coastdown will generate a trip at the low-flow level resulting in earlier RT than the one pump coastdown. The staff observes that the later trip (low-low) results in a more degraded flow without a corresponding decrease in fuel clad heat flux and, therefore, deleterious DNBR. Therefore, the staff finds that the analysis of the single pump coastdown is conservative, since it bounds the reactor dynamics with respect to DNBR.

FSAR Tier 2, Section 15.3 states that the complete loss of flow is assumed to be caused by an electrical power systems fault that affects all four RCPs. The reactor trips at either the low RCP speed setpoint or the low loop flow setpoint. The staff confirmed that these are safety-related signals. The applicant explained that the protective system (PS) logic initiates a RT when two out of the four loops generate a low RCP speed signal.

In order to demonstrate that the necessary functionality is preserved when one PS division (coolant loop) is out of service for maintenance and another fails (single failure), it is assumed that the reactor trips only when the low RCP speed setpoint is reached in all four loops. The staff finds these assumptions conservative. Similarly, the staff concurs that it is conservative to assume the most positive MTC of 0.0 pcm/°C (0.0 pcm /°F), since this maximizes the power. No credit for operator actions was assumed.

The FSAR does not show plots of the components to reactivity and the total reactivity as a function of time during these events. To better understand the system dynamics, in RAI 34,

Question 15.03.03-15.03.04-1, the staff requested that the applicant provide these plots. In an October 31, 2008, response to RAI 34, Question 15.03.03-15.03.04-1, the applicant provided the requested figures that showed plots of the components to reactivity and the total reactivity as a function of time. The staff reviewed the response and found the plots sufficient to resolve staff concerns.. Therefore, the staff considers RAI 34, Question 15.03.03-15.03.04-1resolved.

The plant response to these events and the timely RT is highly dependent on the RCP coastdown. In RAI 34, Question 15.03.03-15.03.04-2, the staff requested that that the applicant quantify the RCP inertia applied for the reactor transients and explain assumptions used in determining the inertia of the RCPs. In RAI 34, Question 15.03.03-15.03.04-3, the staff requested that the applicant provide the RCP coastdown flow characteristics used for determining the mass flow rate in the primary reactor coolant recirculation loops for the reactor transients in FSAR Tier 2, Sections 15.3.1 through 15.3.4. In a September 2, 2008, response to RAI 34, Questions 15.03.03-15.03.04-2 and 15.03.03-15.03.04-3, the applicant stated that the minimum RCS flow rate as a function of time for a four-pump coastdown is controlled as a FSAR Tier 1 item. The flow during four pump coastdown will be confirmed by pre-operational testing. This is acceptable, since the analysis assumptions will be confirmed by test (see FSAR Tier 1, Table 2.2.1-4, "Minimum Flow (% of Initial Flow) During Four Pump Coastdown"). Therefore, the staff considers RAI 34, Questions 15.03.03-15.03.04-2 and 15.03.03-15.03.04-3 jointly resolved by the commitment to flow testing as part of pre-operational testing.

In order to better understand the plant transient response, in RAI 34, Question 15.03.03-15.03.04-4, the staff requested that the applicant explain how the core inlet temperature is computed and used in the determination of DNBR. In a September 2, 2008, response to RAI 34, Question 15.03.03-15.03.04-4, the applicant stated that the core inlet temperature for each of the FSAR Tier 2, Section 15.3 events is computed by S-RELAP5 as the lower plenum fluid temperature. The core inlet temperature, core exit pressure, core inlet mass flux, and fuel rod surface heat flux calculated by S-RELAP5 — in conjunction with the axial power shape that provides the greatest ΔDNBR — are used as a basis for the ΔDNBR computation for the RCP rotor seizure and RCP shaft break. The applicant explained that the complete loss of flow incurs the greatest change in ΔDNBR of any of the transients. Therefore, the complete loss of coolant flow event ΔDNBR (plus margin) forms the basis for the DNBR LCO. Note, that the DNBR LCO less the ΔDNBR during the transient must exceed the DNBR criteria. The staff finds the response acceptable since it describes the conservative means by which the DNBR LCO is determined. Therefore, the staff considers RAI 34, Question 15.03.03-15.03.04-4 resolved.

In order to assess whether the U.S. EPR meets the DNBR acceptance criteria, in RAI 34, Question 15.03.03-15.03.04-5, the staff requested that the applicant provide plots that show the computed DNBR criterion as well as the DNB margin available as a function of time for the reactor transients presented in FSAR Tier 2, Sections 15.3.1 through 15.3.4. In an October 31, 2008, response to RAI 34, Question 15.03.03-15.03.04-5, the applicant stated that, as described in the response to RAI 34, Question 15.03.03-15.03.04-4, the complete loss of forced reactor coolant flow exhibits the greatest change in ΔDNBR. To provide representative DNBR plots and DNBR margin for partial and complete loss of forced reactor coolant flow as a function of time, the MDNBR calculated by the low DNBR channel algorithm is normalized to the DNBR SAFDL. The available margin is represented by the distance from the normalized DNBR line to the 1.0 line. Additionally, the line indicated as "LPD" is the maximum LPD calculated by the high LPD channel algorithm. This value is normalized to either the FCM or cladding strain limits (whichever is more). The response is acceptable to the staff since it provides the requested

information and shows the margin between the plant response and DNBR acceptance criteria. Therefore, the staff considers RAI 34, Question 15.03.03-15.03.04-5 resolved.

In order to better understand the heterogeneity of the core thermal-hydraulic behavior, in RAI 34, Question 15.03.03-15.03.04-6, the staff requested that the applicant provide a comparison between the maximum core local heat flux and the core average heat flux as shown in FSAR Tier 2, Figures 15.3-3, "Partial Loss of Forced Reactor Coolant Flow Event − Core Average Heat Flux," 5.3-7, "Complete Loss of Forced Reactor Coolant Flow Event − Core Average Heat Flux," and 15.3-12, "Reactor Coolant Pump Rotor Seizure Event − Core Average Heat Flux." In an October 31, 2008, response to RAI 34, Question 15.03.03-15.03.04-6, the applicant provided the requested plots and explained that the FSAR Tier 2, Figures 15.3-3, 15.3-7, and 15.3-12 show normalized core average heat flux as a function of time for the partial loss of forced reactor coolant flow, complete loss of forced reactor coolant flow, and RCP rotor seizure events. The core average heat flux for each of the FSAR Tier 2, Section 15.3 events is computed by S-RELAP5. The hot channel heat flux is calculated by multiplying the core average heat flux by the maximum radial peaking factor. The "maximum core local heat flux" presented here is the hot channel heat flux normalized to the core average heat flux at event initiation. The response is acceptable since it provides the requested information and the plots confirm the staff's understanding of the expected plant dynamical behavior. Therefore, the staff considers RAI 34, Question 15.03.03-15.03.04-6 resolved.

In order to assess the DNBR acceptance criteria, in RAI 34, Question 15.03.03-15.03.04-7, the staff requested that the applicant provide plots governing thermal hydraulic quantities that enter into the DNBR calculation and an explanation of the applicability of the DNBR correlation to the range of the relevant parameters. In an October 31, 2008, response to RAI 34, Question 15.03.03-15.03.04-7, the applicant stated:

> For the current U.S. EPR fuel assemblies, there are two CHF correlations: ACH-2 from ANP-10269PA, Revision 0, "The ACH-2 CHF Correlation for the U.S. EPR Topical Report" (Ref. 15-20) and BWU-N from BAW-10199P-A, Revision 0, "The BWU Critical Heat Flux Correlations" (Ref. 15-21).

A table showing the combined range of applicability of both CHF correlations was provided in the October 31, 2008, response to RAI 34, Question 15.03.03-15.03.04-7. The governing parameters for the CHF correlation are pressure, mass velocity, quality, and the geometry where CHF occurs.

The following text from the October 31, 2008, response to RAI 34, Question 15.03.03-15.03.04-7 explains the applicability of each parameter to the CHF correlations:

> Pressure: The minimum and maximum pressure for the Section 15.3 events is 14.803 MPa (2,147 psia) and 16.520 MPa (2,396 psia), respectively. These are within the range of applicability for both the ACH-2 and BWU-N CHF correlations.
>
> Mass Velocity: The maximum mass velocity is not a concern during the Section 15.3 events, because these events consist of a decrease in forced flow. The minimum applicable mass velocity for the ACH-2 is more restrictive than the BWU-N correlation and is conservatively used as the lower limit for licensing analyses. The low range of applicability for the ACH-2 mass velocity is approximately 36 percent of the nominal core inlet flow rate. The core inlet flow

rate does not decrease below 36 percent until well after the RT setpoint is reached. Therefore, applicability of the CHF correlations with respect to the mass velocity is valid for the 15.3 events.

Quality: The maximum exit quality calculated in the limiting Section 15.3 event is less than 22 percent. This is less than the ACH-2 CHF correlation limit of 37 percent.

The applicant has systematically shown the applicability of the DNBR correlation used in the event analysis. Therefore, the staff considers RAI 34, Question 15.03.03-15.03.04-7 resolved.

In RAI 34, Question 15.03.03-15.03.04-8, the staff requested that the applicant provide a plot of the peak fuel centerline temperature as a function of time. In an October 31, 2008, response to RAI 34, Question15.03.03-15.03.04-8, the applicant stated that, as indicated in its October 31, 2008, response to RAI 34, Question 15.03.03-15.03.04-5, the LPD is the maximum LPD calculated by the high LPD channel algorithm. This value is normalized to either the FCM or cladding strain LPD limit, (whichever is more limiting), including all applicable uncertainties. For AOOs, the limiting LPD of cladding strain is more restrictive and is, therefore, used to provide the normalized LPD. The LPD SAFDL is designed to protect both the FCM and cladding strain limits. The plots of LPD as a function of time normalized to the LPD SAFDL are provided as part of the response to RAI 15.03.03-15.03.04-5 for partial and complete loss of forced reactor coolant flow events. From an LPD perspective, RCP rotor seizure or RCP shaft break transient events are similar to the partial and complete loss of forced reactor coolant flow. Since there is no power increase for the RCP rotor seizure or RCP shaft break transient events, no plots of the LPD are provided. There are no fuel failures due to FCM for any of the Section 15.3 transient events. The staff determines the explanation to be acceptable based on its understanding of the system dynamics. Therefore, the staff considers RAI 34, Question 15.03.03-15.03.04-8 resolved.

Loss of Offsite Power

The FSAR states that both LOOP and offsite power-available conditions are considered for each event. LOOP is not considered the limiting single failure, and an additional failure is postulated in the transient and accident analysis. The worst single failure for the loss of flow events is the failure of one protective system division. This conforms to the guidance in RG 1.206 and NUREG-0800. Only the worst condition is shown in the FSAR.

Non-safety-Related Systems Assumed in the Analysis

The non-safety-related PZR spray system is modeled, because it has the potential to make the DNB consequences more severe by limiting the RCS pressure increase during the event. This is conservative, because it would reduce the DNBR margin.

A TT is generated by the check-back signal on a RT. This signal closes the turbine control and stop valves, terminating steam flow to the turbine. Crediting these non-safety-related backup protective systems and components in the DBA analysis following an RT conforms to the regulatory position stated in NUREG-0138 and is acceptable.

15.3.1.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.3.1.6 *Conclusions*

The staff finds that the applicant has identified the most limiting moderate-frequency events, which conforms to the guidance in RG 1.206 and NUREG-0800, Sections 15.3.1 and 15.3.2.

The staff concludes that the most limiting single failure has been assumed in the analysis which conforms to RG 1.53. LOOP has been taken as a consequential failure rather than a single failure, which conforms to the guidance in RG 1.206 and NUREG-0800, Sections 15.3.1 and 15.3.2.

The staff finds that the systems analysis conforms to RG 1.105 in that instrument uncertainties are included in the analytic limit, and complies with GDC 13 in that only safety-related trips are used in the analysis.

The staff concludes that the analysis has used input assumptions and methods to minimize calculated DNBR and maximize primary and secondary system pressure. The staff concludes that the applicant's analysis uses approved methodologies and computer codes.

The staff finds that the analysis shows that there is 95 percent probability/95 percent confidence that the DNBR limit is met.

The staff concludes that the analysis shows that the primary and secondary system pressure does not exceed 110 percent of the design limits.

The staff concludes that the event does not cause a more serious event, and that sufficient safety-related equipment is available to shut down the reactor.

Therefore, the staff finds that:

1. GDC 10 and GDC 20, as they relate to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations including AOOs is met in that the safety analysis shows that acceptance criteria for DNBR and LPD are not exceeded.
2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems is met in that only safety-related instrumentation is used in the analysis.
3. GDC 15, as it relates to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached is met in that the analysis shows that pressure limits are not challenged.
4. GDC 17, as it relates to onsite and offsite electric power systems, is met in that the analysis is performed with and without loss of offsite power.
5. GDC 20, as it relates to the reactor protection system being designed to initiate automatically is met in that the automatic actions are assumed in the safety analysis which shows that SAFDLs are not exceeded.
6. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded is met in that the analysis assumes the most reactive control rod is stuck fully withdrawn.

15.3.2 Complete Loss of Forced Reactor Coolant Flow

Review of this section of the FSAR is documented under Section 15.3.1 of this report.

15.3.3 Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break

This section documents the staff's review of FSAR Tier 2, Sections 15.3.3 and 15.3.4.

15.3.3.1 *Introduction*

The RCP rotor seizure (referred to as a locked rotor) event is a postulated accident resulting in an instantaneous seizure of an RCP rotor. The RCP shaft break event is a postulated accident resulting in a rapid reduction in RCS flow. The sudden decrease in core coolant flow while the reactor is at power degrades core heat transfer and can lead to fuel damage. These events have similar system behavior, are addressed in the same SRP section, and have the same acceptance criteria. Therefore, they are discussed together.

15.3.3.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Sections 15.3.3 and 15.3.4 summarized here, in part, as follows:

The FSAR states that a RCP locked rotor causes flow in the faulted RCS loop to decrease rapidly. The flow decrease results in a full RT when the flow reaches the safety-related, low-low flow setpoint when reactor power is above the protection signal permissive.

The FSAR states that a RCP shaft break will cause flow in the faulted RCS loop to decrease rapidly but at a slower decrease in flow rate compared to the locked rotor seizure event (FSAR Tier 2, Section 15.3.3). A full RT occurs when the flow reaches the safety-related, low-low flow setpoint with reactor power above the protection system permissive. The shaft break also separates the RCP anti-reverse rotation device from the rotor, thereby rendering it ineffective when flow in the affected loop reverses direction.

The applicant's FSAR analysis concludes that the regulatory acceptance criteria are met in that 1) the potential fuel damage is sufficiently limited to allow the core to remain in place and intact with no loss of core cooling capability, and 2) the calculated doses at the site boundary meet the acceptance criteria of 10 CFR Part 100.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 15.3.3 are given in FSAR Tier 2, Chapter 16, Section 3.4.1.

15.3.3.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Sections 15.3.3-15.3.4, "Reactor Coolant Pump

Rotor Seizure and Reactor Coolant Pump Shaft Break," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Sections 15.3.3-15.3.4.

1. GDC 17, as it relates to providing onsite and offsite electric power systems to ensure that structures, systems, and components important to safety will function. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that design conditions of the RCPB are not exceeded and the core is cooled in the event of postulated accidents.

2. GDC 27 and GDC 28, as they relate to the RCS being designed with appropriate margin to ensure that the capability to cool the core is maintained.

3. GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a non-brittle manner and that the probability of propagating fracture is minimized.

4. 10 CFR Part 100, as it relates to the calculated doses at the site boundary.

Acceptance criteria adequate to meet the above requirements include:

1. Identification of which postulated accident is the most limiting.

2. Verification that the plant responds to the transients in such a way that the criteria regarding fuel damage and system pressure are met.

3. Pressure in the reactor coolant and main steam systems should be maintained below acceptable design limits, considering potential brittle as well as ductile failures.

4. The potential for core damage is evaluated on the basis that it is acceptable if the MDNBR remains above the 95/95 DNBR limit for PWRs and the CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see NUREG-0800. Section 4.4). If the DNBR or CPR falls below these values, fuel failure (rod perforation) must be assumed for all rods that do not meet these criteria unless it can be shown, based on an acceptable fuel damage model (see NUREG-0800, Section 4.2), which includes the potential adverse effects of hydraulic instabilities, that fewer failures occur. If rod internal pressure exceeds system pressure, then fuel rods may balloon shortly after entering DNB. The effect of ballooning fuel rods must be evaluated with respect to flow blockage and DNB propagation. Any fuel damage calculated to occur must be of sufficiently limited extent that the core will remain in place and intact with no loss of core cooling capability.

5. A rotor seizure or shaft break in a RCP should not, by itself, generate a more serious condition or result in a loss of function of the RCS or containment barriers.

6. Only safety-grade equipment should be used to mitigate the consequences of the event. Safety functions should be accomplished assuming the worst single failure of a safety system active component. For new applications, LOOP should not be considered a single failure; RCP rotor seizures and shaft breaks should be analyzed with a loss of offsite power in combination with a single active failure.

7. The ability to achieve and maintain long-term core cooling should be verified.

8. This event should be analyzed assuming TT and coincident LOOP and coastdown of undamaged pumps.

9. The analyses of transients are to use an acceptable analytical model, approved methodologies and computer codes. The values of the parameters used in the analytical model should be suitably conservative.

15.3.3.4 *Technical Evaluation*

The staff confirmed that the analysis uses the approved non-loss-of-coolant accident analytical methodology described in AREVA TR ANP-10263P-A and the S-RELAP5 computer code to simulate the event. The change in DNBR is then calculated with the LYNXT core thermal-hydraulic computer code using the RCS response from S-RELAP5 as a boundary condition. The methods used to determine the radiological consequences are described in FSAR Tier 2, Section 15.0.3.8, "Locked Rotor Accident."

The staff confirmed that the analysis was performed with a conservative input including low assumed reactive control rod worth and insertion rate and assumes that the most reactive control rod does not insert into the core, and that insertion of the control rods provides adequate reactivity to bring the RCS to hot shutdown. Soluble boron is not needed to terminate the event. A BOC moderator temperature coefficient of 0.0 pcm/°C (0.0 pcm/°F) is conservatively taken. The worst single failure is failure of one protective division. No credit is taken for operator action.

Following RT and TT, the analysis described in FSAR Tier 2, Section 15.3.2, assumes LOOP. The LOOP trips the remaining RCPs. The staff notes that tripping of the RCPs minimizes heat transfer to the SGs and maximizes primary system pressure. FSAR Tier 2, Section 15.3.2, shows that the primary pressure remains below 110 percent of design and is acceptable.

FSAR Tier 2, Section 15.3.2 shows that the shaft seizure is rapidly terminated by the safety-related low-low flow RT. The trip occurs in less than 2 seconds, and MDNBR occurs shortly thereafter controlled by the reverse flow in the loop with the seized rotor. The heat flux responds slower, controlled by the fuel rod time constant so the MDNBR occurs with near full power heat flux, near normal system pressure, and reduced RCS loop flow.

FSAR Tier 2, Section 15.3.2 states that this trip does not occur soon enough to maintain the DNBR above the 95 percent probability/95 percent confidence DNBR limit. It is conservatively predicted that 8 percent of the core undergoes DNB induced cladding failure. Core cooling capability is maintained. The basis for the value of eight percent fuel damage was not explained and is discussed below.

FSAR Tier 2, Section 15.3.2 states that a RCP shaft break will cause flow in the faulted RCS loop to decrease rapidly but at a slower decrease in flow rate compared to the locked rotor seizure event. Therefore, the shaft seizure bounds the shaft break. The staff concurs with the information provided in FSAR Tier 2, Section 15.3.2, in relation to shaft seizure.

Applicable RAIs and responses for this section are combined and shown in Sections 15.3.1 and 15.3.2 of this report.

In RAI 29, Question 15.03.03-9, the staff requested that the applicant provide an explanation of the analytical and numerical basis for the predicted extent of fuel damage (eight percent) in the RCP shaft seizure postulated accident. In a November 20, 2008, response to RAI 29, Question 15.03.03-9, the applicant stated that a deterministic method is used to predict the extent of fuel damage. This method compares the linear heat generation rate (LHGR) of fuel rods during the RCP rotor seizure event to the LHGR limit that would cause fuel failure due to FCM, excessive uniform clad strain, or DNB. Any fuel rod with an MDNBR below the 95/95 DNBR limit, or exceeding the 95/95 LHGR limits for FCM or clad strain is treated as failed. With these failure thresholds and failure criteria, the number of fuel rods determined to be failed is less than eight percent of the total number in the core. Because there is no significant increase in core power for this event, the FCM design limit is not challenged and no FCM failures are predicted. The staff concluded that the applicant's methodology is conservative. Therefore, the staff considers RAI 29, Question 15.03.03-9 resolved.

In order to assess whether the U.S. EPR meets the DNBR acceptance criteria, in RAI 34, Question 15.03.03-15.03.04-5, the staff requested that the applicant provide plots that show the computed DNBR criterion as well as the DNB margin available as a function of time for the reactor transients presented in FSAR Tier 2, Sections 15.3.1 through 15.3.4. In an October 31, 2008, response to RAI 34, Question 15.03.03-15.03.04-5, the applicant stated that the RCP rotor seizure and broken shaft events are postulated accidents. Analysis of the RCP rotor seizure shows that violation of the DNB SAFDL occurs within the first second of the event initiation. Because computed DNBR is not meaningful after exceeding the critical heat flux, the plots of DNBR as a function of time are not provided. The postulated accident events described in NUREG-0800, Sections 15.3.3 and 15.3.4, RCP rotor seizure and shaft break events, are evaluated with a conservative fuel failure analysis. The RCP rotor seizure event has more severe flow degradation than the broken shaft event. Therefore, the most limiting rotor seizure event is analyzed for fuel failure. A transient LYNXT model, including uncertainties described in FSAR Tier 2, Section 4.4.4.5.2, "Core Analysis," is used for a conservative evaluation of the DNBR SAFDL. Rod failure by DNB is evaluated by comparing the rod average linear heat generation rate of a given fuel rod to the minimum rod average LHGR required for DNB. Evaluation of the most limiting fuel rod internal pressure shows that fuel failures do not propagate to the surrounding rods in an assembly. It is determined that a total of eight percent of the fuel rods experience fuel failure due to DNB. The radiological consequences are assessed based on the eight percent of fuel failure and reported in Section 15.0.3 of this report. The response is acceptable since it explains using accepted methods how the fraction of fuel failure is conservatively predicted. Therefore, the staff considers RAI 34, Question 15.03.03-15.03.04-5 resolved.

15.3.3.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.3.3.6 *Conclusions*

The staff finds that the applicant has identified those postulated accidents which conform to the guidance in RG 1.206 and NUREG-0800, Sections 15.3.3 and 15.3.4.

The staff finds that the most limiting single failure has been assumed in the analysis which conforms to RG 1.53. LOOP has been taken as a consequential failure rather than a single failure in conformance with the guidance in RG 1.206 and NUREG-0800, Sections 15.3.1

and 15.3.2. The staff finds that the systems analysis conforms to RG 1.105 in that instrument uncertainties are included in the analytic limit, and complies with GDC 13 in that only safety-related trips are used in the analysis.

The staff finds that the analysis has used input assumptions and methods to minimize calculated DNBR and maximize primary and secondary system pressure, and that the analysis uses approved methodologies and computer codes.

The staff finds that the analysis of the RCP shaft seizure shows that the consequences of this postulated accident are acceptable. The staff concludes that fuel damage is less than eight percent based on a conservative calculation and the radiological consequences of the event are described in FSAR Tier 2, Section 15.0.3.8.2, "Input Parameters and Initial Conditions."

The staff concludes that the RCP shaft break is less severe than the RCP shaft seizure, since the shaft break results in a free wheeling pump rotor and corresponding flow coastdown, while the shaft seizure results in an instantaneous loss of flow. As such, the staff concludes that the RCP shaft break is bounded by the RCP shaft seizure.

The staff finds that the analysis shows that the primary and secondary system pressure does not exceed 110 percent of the design limits.

The staff concludes that the event does not cause a more serious event, and the reactor can be shut down using only safety-related equipment.

Therefore, the staff finds that:

1. GDC 17, as it relates to providing onsite and offsite electric power systems is met in that the safety analysis was performed with the more limiting assumption of loss of offsite power.

2. GDC 27 and GDC 28, as they relate to the RCS being designed with appropriate margin to ensure that the capability to cool the core is maintained are met in that the analysis shows that fuel failures are not propagating and, therefore, does not challenge core cooling.

3. GDC 31, as it relates to the RCS being designed with sufficient margin to ensure that the boundary behaves in a non-brittle manner and that the probability of propagating fracture is minimized is met in that the analysis shows that pressure limits are not exceeded and that the event does not cause a rapid vessel cooldown.

4. 10 CFR Part 100, as it relates to the calculated doses at the site boundary is discussed in Section 15.0.3 of this report.

15.4 Reactivity and Power Distribution Anomalies

Core reactivity and power distribution within the core are controlled for the safe and efficient operation of the reactor. Several AOOs and postulated accidents can alter the core reactivity and power distribution and disrupt reactor operation. Core reactivity changes can result from boron concentration changes, overcooling of the RCS, the addition of cooler water to the RCS, spurious control rod motion, or a control rod ejection from the core region. Power distribution

changes can result from control rod movement, control rod ejection, or location of fuel assemblies in an incorrect position. FSAR Tier 2, Section 15.4, "Reactivity and Power Distribution Anomalies," describes the following events and includes analyses that demonstrate that acceptance criteria are met in these subsections:

- FSAR Tier 2, Section 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low-Power Startup Condition"
- FSAR Tier 2, Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power"
- FSAR Tier 2, Section 15.4.3, "Control Rod Misoperation (System Malfunction or Operator Error)"
- FSAR Tier 2, Section 15.4.4, "Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature"
- FSAR Tier 2, Section 15.4.5, "Flow Controller Malfunction Causing an Increase in Boiling Water Reactor Core Flow Rate" (not applicable to the U.S. EPR)
- FSAR Tier 2, Section 15.4.6, "Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant"
- FSAR Tier 2, Section 15.4.7, "Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position"
- FSAR Tier 2, Section 15.4.8, "Spectrum of Rod Ejection Accidents in a Pressurized Water Reactor"
- FSAR Tier 2, Section 15.4.9, "Spectrum of Rod Drop Accidents (BWR)" (not applicable to the U.S. EPR)

Transient and accident categorization and classification are discussed in FSAR Tier 2, Section 15.0.0. The computer codes used to analyze DBAs are discussed in FSAR Tier 2, Section 15.0.2. Radiological consequences for DBAs are discussed in FSAR Tier 2, Section 15.0.3.

15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low-Power Startup Condition

15.4.1.1 *Introduction*

The uncontrolled control rod assembly withdrawal from a subcritical or low-power startup condition event is defined as an uncontrolled addition of reactivity due to the withdrawal of banks of rod cluster control assemblies (RCCAs) at hot shutdown or hot standby conditions. This event is an AOO of moderate frequency.

15.4.1.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.4.1, summarized here, in part, as follows:

FSAR Tier 2, Section 15.4 states that the uncontrolled rod control cluster assembly (RCCA) event is only analyzed during the startup and at power modes of reactor operation. Reactor operations in hot standby, cold shutdown, and the refueling mode preclude this event, because the RT breakers are not energized and the RCS boron concentration is sufficient to preclude return to criticality even with all the RCCAs fully withdrawn.

For startup and power operations, an uncontrolled addition of reactivity to the reactor core by an uncontrolled RCCA bank withdrawal leads to a power excursion. The neutron flux response to the continuous reactivity insertion is characterized by a fast rise limited by the reactivity feedback effect of the negative fuel temperature coefficient. If the detected neutron flux exceeds the protections system setpoints, an RT is initiated.

FSAR Tier 2, Section 15.4 states that the applicable acceptance criteria for the uncontrolled RCCA bank withdrawal from a subcritical or low-power condition event used in the FSAR analysis are:

- The DNBR thermal margin limit is met.
- Fuel centerline temperatures do not exceed the melting point.
- Uniform cladding strain does not exceed one percent.

The applicant's FSAR Tier 2, Section 15.4 analysis concludes that the regulatory acceptance criteria are met, and there are no radiological consequences or releases from this event.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 15.4.1 are given in FSAR Tier 2, Chapter 16, Section 3.2.3.

15.4.1.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.4.1, "Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.4.1.

1. GDC 10, which requires that SAFDLs are not to be exceeded during normal operation, including the effects of AOOs.
2. GDC 13, which requires that the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.
3. GDC 17, which requires provision of an onsite electric power system and an offsite electric power system to permit functioning of structures, systems, and components important to safety.

4. GDC 20, which requires that the protection system initiate automatically appropriate systems to assure that SAFDLs are not exceeded as a result of AOOs.

5. GDC 25, which requires that the reactor protection system be designed to assure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control system.

Acceptance criteria adequate to meet the above requirements include:

1. The thermal margin limits for DNBR are met.

2. Fuel centerline temperatures do not exceed the melting point.

15.4.1.4 *Technical Evaluation*

The staff confirmed that the analysis of the uncontrolled control rod assembly withdrawal from a subcritical or low-power event uses the approved non-loss-of-coolant accident analytical methodology described in AREVA TR ANP-10263P-A and the S-RELAP5 computer code to simulate the event. The change in DNBR is then calculated with the LYNXT core thermal-hydraulic computer code using the RCS response from S-RELAP5 as a boundary condition.

S-RELAP5 includes a point kinetics reactivity model, which the staff finds conservative to model this event since credit is not taken for local reactivity changes to control or terminate the event.

The staff finds that the analysis is performed with a conservatively positive moderator temperature coefficient, conservative Doppler temperature coefficient, bounding delayed neutron fraction, and conservative scram reactivity including the assumed failure of the most reactive control rod to insert and conservatively small scram reactivity including the assumed failure of the most reactive control rod to insert (FSAR Tier 2, Table15.0-6). The staff observes that these conditions maximize the power excursion during the event. The transient is terminated by safety-grade signals: High positive neutron flux rate of change, low doubling time of the intermediate range neutron flux, and intermediate range high neutron flux.

FSAR Tier 2, Section 15.4 shows a TT follows an RT and that LOOP is assumed to occur at TT. The staff observes that the LOOP causes coastdown of the RCPs minimizing coolant loop flow and heat transfer to the SGs and maximizes primary system pressure. The LOOP is not taken as the independent single failure in conformance with the guidance in RG 1.206 and NUREG-0800. The staff concurs that the worst single failure is taken as failure of one protective division.

In order to assess conformance with acceptance criteria for this transient, in RAI 29, Question 05.04.01-1, the staff requested that the applicant provide plots of DNBR and peak fuel centerline temperature as a function of time during this event. In a September 2, 2008, response to RAI 29, Question 05.04.01-1, the applicant provided plots of the MDNBR and peak fuel centerline temperature normalized to their respective SAFDLs. The figures show margin to the limits (Reference FSAR Tier 2, Figure 15.4-54, “Uncontrolled Control Bank Withdrawal at Power − Representative Plot of Normalized Minimum DNBR and Maximum LPD to SAFDL”).

Because this event is initiated from a low power condition where the in-core monitoring system is inactive, a deterministic evaluation of DNBR and FCM is performed with the thermal-hydraulic

code LYNXT. Additionally, a deterministic calculation of cladding strain, using the COPERNIC2 code, for this event shows a maximum cladding strain of 0.80 percent. These calculations use approved methods and are acceptable. Therefore, the staff considers RAI 29, Question 05.04.01-1 resolved.

In RAI 29, Question 15.04.01-2, the staff requested that the applicant provide clarification of the labeling of data in certain figures contained in FSAR Tier 2, Section 15.4. In an October 1, 2008, response to RAI 29, Question 15.04.01-2, the applicant provided the requested clarification of the figures and suggested modifications to the FSAR to address staff concerns. The staff reviewed the figures and found that they clearly show the parametric behavior of variables as a function of time in each reactor loop. The staff confirmed that the modified figures were contained in Revision 1 of the FSAR. Therefore, the staff considers RAI 29, Question 05.04.01-2 resolved.

15.4.1.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.4.1.6 *Conclusions*

The staff finds that the applicant has classified the uncontrolled control rod assembly withdrawal from a subcritical or low-power startup condition as a moderate-frequency event in conformance with the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the analysis has used conservative input assumptions and methods to minimize calculated DNBR and maximize primary pressure and maximize fuel temperature.

The staff concludes that the analysis shows that there is 95 percent probability/95 percent confidence that the DNBR limit is met.

The staff concludes that the analysis shows that fuel centerline temperatures do not exceed the melting point.

The staff finds that the most limiting single failure has been assumed in the analysis in conformance with RG 1.53. The staff concludes that LOOP has been taken as a consequential failure rather than a single failure in conformance with the guidance in RG 1.206 and NUREG-0800.

The staff finds that the systems analysis conforms to RG 1.105 in that instrument uncertainties are included in the analytic limit, and complies with GDC 13 in that only safety-related trips are used in the analysis.

The staff finds that the event does not cause a more serious event, and that sufficient safety-related equipment is available to shut down the reactor.

The staff concludes that analyses were performed by the applicant with reviewed and approved codes and methods.

Therefore, the staff finds that:

1. GDC 10, which requires that SAFDLs are not to be exceeded during normal operation, including the effects of AOOs is met in that the analysis shows that DNBR, and fuel centerline temperature limits are met.

2. GDC 13, which requires that the availability of instrumentation is met in that only safety-related instrumentation is assumed available in that the analysis.

3. GDC 17, which requires provision of an onsite electric power system and an offsite electric power system, is met in that the analysis assumes the more limiting loss of off-site power.

4. GDC 20, which requires that the protection system initiate automatically appropriate systems to assure that SAFDLs are not exceeded, is met in that the analysis takes credit for only the safety-related reactor protection system and associated automatic actuation.

5. GDC 25, which requires that the reactor protection system be designed to assure that SAFDLs are not exceeded, is met in that the analysis assumed a single malfunction of the reactivity control system.

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

15.4.2.1 *Introduction*

The uncontrolled control rod assembly withdrawal at power event is defined as an uncontrolled addition of reactivity due to the withdrawal of rod control cluster assembly banks during power operation. This AOO event may result from either a failure in the RCCA position control system or from an operator error.

15.4.2.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," summarized here, in part, as follows:

The FSAR states that the uncontrolled withdrawal of an RCCA bank causes an increase in core power with a corresponding increase in heat flux, reactor coolant temperature, and pressure. Depending on the power level and point in the fuel cycle, the transient terminates on RT from one of several RT protection system signals. The at-power event can result in either a slow or a fast increase in reactivity. In a slow reactivity increase, the increase in coolant temperature follows the increase in reactor power. In a fast reactivity increase, the reactor power increases at a much faster rate than the coolant temperature. The slow reactivity transient is terminated by a high pressurizer level, low DNBR, high core power level, or high SG pressure trips. The fast reactivity transient is terminated by the high neutron flux rate of change RT. In addition to the rate of reactivity increase, the limiting transients for the RCCA bank withdrawal at power depend on the starting reactor power level and the point in the fuel cycle burn up. The analyses considered various power levels and fuel cycle burn up in order to determine and bound the limiting case.

FSAR Tier 2, Section 15.4.2 states that the applicable acceptance criteria for the uncontrolled RCCA bank withdrawal from power event used in the analysis are:

- The DNBR thermal margin limit is met.
- Fuel centerline temperatures do not exceed the melting point.
- Uniform cladding strain does not exceed one percent.

The applicant's analysis concludes that these criteria are met, and there are no radiological consequences or releases from this event.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 15.4.2 are given in FSAR Tier 2, Chapter 16, Sections 3.2.3 and 3.4.1.

15.4.2.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.4.2, "Uncontrolled Control Rod Assembly Withdrawal at Power," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.4.2.

1. GDC 10, which requires that SAFDLs are not to be exceeded during normal operation, including the effects of AOOs.
2. GDC 13, which requires the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.
3. GDC 17, which requires provision of an onsite electric power system and an offsite electric power system to permit functioning of structures, systems, and components important to safety.
4. GDC 20, which requires that the protection system initiate automatically appropriate systems to assure that SAFDLs are not exceeded as a result of AOOs.
5. GDC 25, which requires that the reactor protection system be designed to assure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control system.

Acceptance criteria adequate to meet the above requirements include:

1. The thermal margin limits for DNBR are met.
2. Fuel centerline temperatures do not exceed the melting point.

15.4.2.4 *Technical Evaluation*

Methods and Codes

The staff confirmed that the analysis of the uncontrolled control rod assembly withdrawal at power event uses the approved non-loss-of-coolant accident analytical methodology described in AREVA TR ANP-10263P-A and the S-RELAP5 computer code to simulate the event.

The applicant's implementation of S-RELAP5 includes a point kinetics reactivity model, which the staff finds conservative to model this event since credit is not taken for local reactivity changes to control or terminate the event.

The staff finds that the low DNBR RT employs an online dynamic setpoint methodology described in AREVA TR ANP-10287P. The staff review of the online dynamic setpoint methodology is documented in the staff's SER on ANP-10287P.

The staff notes that the analysis was performed at 100 percent, 60 percent, and 25 percent of full power with a bounding control rod differential worth.

The staff finds that the analysis is performed with a conservatively bounding moderator temperature coefficient, conservative Doppler temperature coefficient, and bounding delayed neutron fraction, and conservatively small scram reactivity including the assumed failure of the most reactive control rod to insert (FSAR Tier 2, Table15.0-6). The staff notes that these assumptions maximize the power excursion during the event. The staff concurs that the worst single failure is failure of one protective system division.

The staff notes that the transient is terminated by safety-grade signals: Low DNBR; high LPD; high neutron flux rate-of-change; high core power level; high pressurizer level; and high SG pressure. The staff confirmed that the assumptions in the analyses are selected to conservatively minimize DNB margin by assuming that the pressure control system is operational, pressurizer spray flow is at its maximum value, and the PSRV opening and closing setpoints are biased low. The staff observes that the pressurizer spray reduces primary pressure and in combination with increasing primary temperature minimizes DNBR margin.

FSAR Tier 2, Section 15.4.2 states that a TT follows an RT, and that LOOP is assumed to occur at TT. The staff observes that the LOOP causes coastdown of the RCPs minimizing coolant loop flow and heat transfer to the SGs and maximizes primary system pressure. The LOOP is not taken as the independent single failure in conformance with the guidance in RG 1.206 and NUREG-0800.

In order to assess conformance with the acceptance criteria for this transient, in RAI 29, Question 05.04.02-1, the staff requested that the applicant provide plots of DNBR and peak fuel centerline temperature as a function of time during this event. In an October 1, 2008, response to RAI 29, Question 15.04.02-1, the applicant provided the requested plots of the MDNBR and peak fuel centerline temperature normalized to their respective SAFDLs. The figures show margin to the limits (Reference FSAR Tier 2, Figure 15.4-47, "Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature − Normalized Minimum DNBR and Maximum LPD to SAFDL"). A representative plot of the MDNBR and maximum LPD, normalized to their respective SAFDLs, is shown. This event results in both low DNBR channel and high LPD channel in-core trips. (See response to RAI 34, Question 15-2 in Section 15.0.1.4 of this report.) The response is acceptable since the requested plots are provided and that the plots

show margin to the acceptance criteria. Therefore, the staff considers RAI 29, Question 05.04.02-1 resolved.

In order to better understand the event, in RAI 29, Question 15.04.02-2 the staff requested that the applicant provide the scram reactivity used in the analysis of the "Uncontrolled Control Bank Withdrawal at Power." In a September 2, 2008, response to RAI 29, Question 15.04.02-2, the applicant provided the scram reactivity at 100 percent, 60 percent, and 25 percent of full power. These values conform to information shown throughout Chapter 15 and are acceptable. Therefore, the staff considers RAI 29, Question 15.04.02-2 resolved.

The transient is terminated by safety-grade signals: Low DNBR; high LPD; high neutron flux rate-of-change; high core power; high pressurized level; and high SG pressure. The assumptions in the analyses are selected to conservatively minimize DNB margin by assuming that the pressure control system is operational, pressurizer spray flow is at its maximum value, and the PSRV opening and closing setpoints are biased low. Pressurizer spray reduces primary pressure and in combination with increasing primary temperature minimizes DNBR margin.

FSAR Tier 2, Section 15.4.2 states that a TT follows an RT, and that LOOP is assumed to occur at TT. The LOOP causes coastdown of the RCPs minimizing coolant loop flow and heat transfer to the SGs and maximizes primary system pressure. The LOOP is not taken as the independent single failure in conformance with the guidance in RG 1.206 and NUREG-0800.

FSAR Tier 2, Section 15.4.2 states that:

> Neither a single failure nor equipment taken out of service for maintenance makes this event more severe. The protection system is the only system or equipment that mitigates this event and its redundant design makes it single-failure proof.

The staff agrees with the applicant's assessment since no other single failure will affect the course of this event.

15.4.2.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.4.2.6 *Conclusions*

The staff finds that the applicant has classified the uncontrolled control rod assembly withdrawal at power event as a moderate-frequency event, which conforms to the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the analysis has used conservative input assumptions and methods to minimize calculated DNBR and maximize fuel temperature.

The staff concludes that the analysis shows that there is 95 percent probability/95 percent confidence that the DNBR limit is met.

The staff concludes that the analysis shows that fuel centerline temperatures do not exceed the melting point.

The staff finds that the most limiting single failure has been assumed in the analysis in conformance with RG 1.53, and that LOOP has been taken as a consequential failure rather than a single failure in conformance with the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the event does not cause a more serious event, and that sufficient safety-related equipment is available to shut down the reactor.

The staff concludes that analyses were performed with reviewed and approved codes and methods.

Therefore, the staff finds that:

1. GDC 10, which requires that SAFDLs are not to be exceeded during normal operation, including the effects of AOOs is met in that the analysis shows that DNBR, and fuel centerline temperature limits are met.

2. GDC 13, which requires that the availability of instrumentation is met in that only safety-related instrumentation is assumed available in that the analysis.

3. GDC 17, which requires provision of an onsite electric power system and an offsite electric power system, is met in that the analysis assumes the more limiting loss of off-site power.

4. GDC 20, which requires that the protection system initiate automatically appropriate systems to assure that SAFDLs are not exceeded, is met in that the analysis takes credit for only the safety-related reactor protection system and associated automatic actuation.

5. GDC 25, “Protection System Requirements for Reactivity Control Malfunctions,” which requires that the reactor protection system be designed to assure that SAFDLs are not exceeded, is met in that the analysis assumed a single malfunction of the reactivity control system.

15.4.3 Control Rod Misoperation (System Malfunction or Operator Error)

15.4.3.1 *Introduction*

This section presents the analysis of three types of postulated rod control cluster assembly misoperation events, in the following FSAR Tier 2, Section 15.4.3 sub-sections:

- Section 15.4.3.1, “Dropped RCCA or RCCA Sub-Bank”
- Section 15.4.3.2, “Statically Misaligned RCCA”
- Section 15.4.3.3, “Single RCCA Withdrawal”

These events are classified as AOOs. These events are discussed together, since there is a common SRP section and a common set of acceptance criteria. Results and conclusions are presented for the most limiting cases.

15.4.3.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.4.3, summarized here, in part, as follows:

Dropped RCCA or RCCA Sub-Bank

The FSAR states that the dropped RCCA or RCCA sub-bank event is initiated by de-energizing a drive mechanism by a RCCA malfunction during power operation, or by malfunction of the control system for the sub-bank or rod bank. The malfunction results in a single RCCA, sub-bank or complete control bank, dropping into the core. Insertion of the affected RCCAs decreases the reactor power and increases the radial power peaking in the reactor core. RCS temperature initially decreases in response to the decrease in reactor power. Power then increases due to reactivity feedback and automatic withdrawal of other control banks. The magnitude of this increase depends on the operating mode of the plant and the protection system response. The plant response to a dropped RCCA (single RCCA, sub-bank, or full bank) is further dependent on the mode of operation of the RCCA system and the turbine control valve. The analysis presented in the FSAR concluded that the plant instrumentation and protection systems functions are sufficient to preclude fuel or cladding damage and that the core is adequately cooled during the dropped RCCA event.

Statically Misaligned RCCA

The effect of a statically misaligned RCCA is a local distortion of the neutron flux and local power. Protection is provided by the in-core instrumentation and setpoint methodology as discussed in AREVA TR ANP-10287P.

Single RCCA Withdrawal

The accidental withdrawal of a single RCCA at power can occur while operating the RCCAs in manual due to operator error or due to electrical or mechanical failures. The withdrawal of a single RCCA causes an insertion of reactivity and, therefore, an increase in average core power and temperature. Local power peaking occurs in the zone where the RCCA is located. The analysis presented in the FSAR concluded that the RCS response to the withdrawal of a single RCCA at power event is bounded by the RCS response of the withdrawal of an RCCA bank event at power discussed in FSAR Tier 2, Section 15.4.2.

FSAR Tier 2, Section 15.4.3 states that the applicable acceptance criteria for the three misoperation of control rod event scenarios used in the FSAR analysis are:

- The DNBR thermal margin limit is met.
- Fuel centerline temperatures do not exceed the melting point.
- Uniform cladding strain does not exceed one percent.

The applicant's FSAR analysis concludes that these criteria are met, and there are no radiological consequences or releases from this event.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 15.4.3 are given in FSAR Tier 2, Chapter 16, Sections 3.1.4, "Rod Control Cluster Assembly Group Alignment Limits"; 3.1.5, "Shutdown Bank Insertion Limits"; 3.1.7, "Rod Control Cluster Assembly (RCCA) Position Indication"; and 3.2.3.

15.4.3.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.4.3, "Control Rod Misoperation (System Malfunction or Operator Error)," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.4.3.

1. GDC 10, which requires that SAFDLs are not to be exceeded during normal operation, including the effects of AOOs.

2. GDC 13, which requires the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 20, which requires that the protection system initiate automatically appropriate systems to assure that SAFDLs are not exceeded as a result of AOOs.

4. GDC 25, which requires that the reactor protection system be designed to assure that SAFDLs are not exceeded in the event of a single malfunction of the reactivity control system.

Acceptance criteria adequate to meet the above requirements include:

1. The thermal margin limits for DNBR are met.

2. Fuel centerline temperatures do not exceed the melting point.

15.4.3.4 *Technical Evaluation*

The staff confirmed that the analysis of the dropped RCCA or RCCA sub-bank, statically misaligned RCCA, and single RCCA withdrawal use the approved non-loss-of-coolant accident analytical methodology described in AREVA TR ANP 10263P-A and the S-RELAP5 computer code to simulate the event. S-RELAP5 includes a point kinetics reactivity model, which is adequate to model this event, since point kinetics overestimates the transient excursion relative to spatial kinetics. The low DNBR RT employs an online dynamic setpoint methodology described in AREVA TR ANP-10287P.

FSAR Tier 2, Section 15.4.3 shows that the dropped control rod or dropped sub-bank is protected by the safety-related low DNBR RT. Events were analyzed at BOC, EOC, with corresponding MTC and Doppler coefficients, and limiting (maximum and minimum) values of dropped rod and bank reactivity worths. The analyses were performed with the ACT in both automatic and manual mode (in manual mode the ACT remains as is and the average coolant temperature is taken as a fixed initial condition). The staff concludes that these analyses are

performed over a range of conditions to ensure that the online DNBR trip setpoints are adequate.

FSAR Tier 2, Section 15.4 shows a TT follows an RT and that LOOP is assumed to occur at TT. The staff observes that the LOOP causes coastdown of the RCPs minimizing coolant loop flow and heat transfer to the SGs and maximizes primary system pressure. The LOOP is not taken as the independent single failure in conformance with the guidance in RG 1.206 and NUREG-0800. Failure of one protective division is taken as the single random failure in the analysis for the uncontrolled RCCA withdrawal from subcritical, uncontrolled bank withdrawal at power, single RCCS withdrawal at power, and RCCS misalignment. Failure of the highest ex-core signal input to the control rod drive control system (CRDCS) is taken as the single failure in the RCCS drop analysis. The staff concurs that this is the limiting single failure since this results in a delayed RT and a corresponding decrease in DNBR.

In order to assess conformance with acceptance criteria for this transient, in RAI 29, Question 15.04.03-1, the staff requested that the applicant provide plots of DNBR and peak fuel centerline temperature as a function of time during this event. In a November 20, 2008, response to RAI 29, Question 15.04.03-1, the applicant stated: The DNBR normalized to the DNBR SAFDL as a function of time is provided in new FSAR Tier 2, Figure 15.4-55, "Dropped RCCA - Minimum DNBR and Maximum LPD Normalized to SAFDL." The figure is provided in Revision 1 of FSAR Tier 2, Section 15.4. The low DNBR RT provides adequate protection of the DNBR SAFDL for the bank drop event. The LPD SAFDL is a criterion for preventing fuel centerline melting and excessive cladding strain (see FSAR Tier 2, Section 15.0.0.3.9, "Overview of the Incore Transient Methodology"). The maximum LPD normalized to the LPD SAFDL as a function of time is provided in new FSAR Tier 2, Figure 15.4-55. Because the LPD SAFDL is not exceeded, fuel centerline melting and excessive cladding strain do not occur. Examination of the new figures shows the expected trend with time and margin to the limits. The staff finds the response acceptable since the requested plots are provided and that the plots show margin to the acceptance criteria. Therefore, the staff considers RAI 29, Question 15.04.03-1 resolved.

15.4.3.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.4.3.6 *Conclusions*

The staff finds that the applicant has classified the dropped RCCA or RCCA sub-bank, statically misaligned RCCA, and single RCCA withdrawal as moderate-frequency events, which conforms to the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the analysis has used conservative input assumptions and methods to minimize calculated DNBR and maximize primary and maximize fuel temperature.

The staff concludes that the analysis shows that there is 95 percent probability/95 percent confidence that the DNBR limit is met.

The staff concludes that the analysis shows that fuel centerline temperatures do not exceed the melting point.

The staff finds that the most limiting single failure has been assumed in the analysis in conformance with RG 1.53, and that LOOP has been taken as a consequential failure rather than a single failure which conforms to the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the systems analysis conforms to RG 1.105 in that instrument uncertainties are included in the analytic limit, and complies with GDC 13 in that only safety-related trips are used in the analysis.

The staff concludes that the event does not cause a more serious event, and that sufficient safety-related equipment is available to shut down the reactor.

The staff concludes that the analyses were performed by the applicant with reviewed and approved codes and methods.

Therefore, the staff finds that:

1. GDC 10, which requires that SAFDLs are not to be exceeded during normal operation, including the effects of AOOs is met in that the analysis shows that DNBR, and fuel centerline temperature limits are met.

2. GDC 13, which requires that the availability of instrumentation is met in that only safety-related instrumentation is assumed available in that the analysis.

3. GDC 20, which requires that the protection system initiate automatically appropriate systems to assure that SAFDLs are not exceeded, is met in that the analysis takes credit for only the safety-related reactor protection system and associated automatic actuation.

4. GDC 25, which requires that the reactor protection system be designed to assure that SAFDLs are not exceeded, is met in that the analysis assumed a single malfunction of the reactivity control system.

15.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

15.4.4.1 *Introduction*

This event is an AOO initiated from a condition in which the plant has undergone a partial RT due to the coastdown of one RCP.

15.4.4.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.4.4, summarized here, in part, as follows:

The FSAR states that the loss of a RCP at power reduces reactor power to approximately 50 percent. The time the reactor can operate at reduced power without all four RCPs running is limited by the Technical Specifications. The idle RCP must be restarted within the required time or the reactor must be shut down in accordance with the applicable Technical Specification(s). The most significant impact of restarting the RCP while the reactor is at part power is the rapid

increase in core flow. This rapid increase in flow rate decreases the average temperature of the RCS cold-leg flow to the core resulting in increasing reactivity and core power. This event returns the plant to full flow operation and the operator can then increase core power level to full power. The protection system does not activate unless core protection limits are challenged. The PS is designed to terminate this transient before the DNB limit is reached.

FSAR Tier 2, Section 15.4 shows the following applicable acceptance criteria for this event:

- MDNBR remains above the design limit.
- Peak RCS pressure does not exceed 110 percent of design pressure.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 15.4.4 are given in FSAR Tier 2, Chapter 16, Sections 3.2.3, 3.4.1, and 3.4.4.

15.4.4.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Sections 15.4.4-15.4.5, "Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Sections 15.4.4-15.4.5.

1. GDC 10 and GDC 20, as they relate to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations and AOOs.
2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.
3. GDC 15 and GDC 28, as they relate to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached during normal operations and AOOs.
4. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded during AOOs. This is accomplished by ensuring that the analysis accounts for appropriate margins for malfunctions, such as stuck rods.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
2. Fuel-cladding integrity shall be maintained by ensuring that the MDNBR remains above the 95 percent probability/95 percent confidence DNBR limit for PWRs.
3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

4. The requirements stated in RG 1.105 are used with regard to their impact on the plant response to the type of AOOs addressed in NUREG-0800.

5. RG 1.53, as it relates to identifying and assuming in the analysis the most limiting plant systems single failure, as defined in the "Definitions and Explanations" section of 10 CFR Part 50, Appendix A.

6. The guidance provided in SECY-77-439, "Single Failure Criterion," August 17, 1977; SECY 94-084, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs," March 28, 1994; and RG 1.206 with respect to the consideration of the performance of non-safety-related systems during transients and accidents, as well as the consideration of single failures of active and passive systems (especially as they relate to the performance of check valves in passive systems), must be evaluated and verified.

15.4.4.4 *Technical Evaluation*

The staff confirmed that the analysis of the startup of an inactive RCP at an incorrect temperature event uses the approved non-loss-of-coolant accident analytical methodology described in AREVA TR ANP-10263P-A and the S-RELAP5 computer code to simulate the event. A deterministic evaluation of DNBR is performed with the thermal-hydraulic code LYNXT. S-RELAP5 includes a point kinetics reactivity model, which is conservative to model this event since point kinetics overestimates the transient excursion relative to spatial kinetics. A DNBR trip is not predicted to occur during the startup of the RCP. However the online dynamic setpoint methodology (in-core trip system) provides protection for DNBR and LPD and, therefore, fuel centerline temperature and cladding strain limits. AREVA TR ANP-10287P describes the methodology.

The analysis of the startup of an inactive reactor coolant pump at an incorrect temperature does not predict a RT. The staff notes that this conforms to the analysis, which shows a dynamic increase of power of about 15 percent, while the flow increases by 25 percent so that the margin to DNBR increases.

The staff finds that the analysis is conservatively performed with a bounding negative MTC and Doppler coefficient since this maximizes the power excursion. The staff concurs that the analysis is conservatively performed with the pressurizer spray assumed to be on, which is consistent with the startup of the fourth RCP, as is the availability of power to the RCPs throughout the event.

In order to assess conformance with acceptance criteria for this transient, in RAI 29, Question 15.04.04-15.04.05-1, the staff requested that the applicant provide plots of DNBR and peak fuel centerline temperature as a function of time during this event. In an October 1, 2008, response to RAI 29, Question 15.04.04-15.04.05-1, the applicant provided plots showing margin to the limits. The figures are shown in Revision 1 of the FSAR Tier 2, Section 15.4. They illustrate that the time scale of the increase in power due to decrease in coolant temperature as the fourth RCP starts up and the increase in flow is such that DNBR margin is preserved. Therefore, the staff considers RAI 29, Question 15.04.04-15.04.05-1 resolved.

15.4.4.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.4.4.6 *Conclusions*

The staff finds that the applicant has classified the startup of an inactive reactor coolant pump at an incorrect temperature as a moderate-frequency event, which conforms to the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the analysis has used conservative input assumptions and methods to minimize calculated DNBR and maximize primary and maximize fuel temperature.

The staff concludes that the analysis shows that there is 95 percent probability/95 percent confidence that the DNBR limit is met.

The staff concludes that the analysis shows that fuel centerline temperatures do not exceed the melting point.

The staff finds that the most limiting single failure has been assumed in the analysis which conforms to RG 1.53, and that LOOP has been taken as a consequential failure rather than a single failure in conformance with the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the event does not cause a more serious event, and that sufficient safety-related equipment is available to shut down the reactor.

The staff concludes that the analyses were performed with reviewed and approved codes and methods.

Therefore, the staff finds that:

1. GDC 10 and GDC 20, as they relate to the RCS being designed with appropriate margin to ensure that SAFDLs are not exceeded during normal operations and AOOs is met in that the safety analysis shows that fuel limits are not exceeded and the RCS pressure does not exceed design limits.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges is met in that the safety analysis takes credit only for safety-related instrumentation.

3. GDC 15 and GDC 28, as they relate to the RCS and its associated auxiliaries being designed with appropriate margin to ensure that the pressure boundary will not be breached, is met in that the safety analysis shows that pressure limits are not exceeded.

4. GDC 26, as it relates to the reliable control of reactivity changes to ensure that SAFDLs are not exceeded during AOOs is met in that the safety analysis only takes credit for the safety-related RPS, and that appropriate margins for malfunctions, such as stuck rods, are accounted for in the safety analysis.

15.4.5 Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

This event is not applicable to the U.S. EPR.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant (Inadvertent Decrease in Boron Concentration in the Reactor Coolant System)

15.4.6.1 *Introduction*

This section presents the analysis of the inadvertent boron dilution event, which is classified as an AOO. The principal scenarios leading to a potential uncontrolled boron dilution event are as follows:

- Operator addition of deborated water to the RCS
- Malfunction in the coolant degasification system
- Malfunction in the coolant purification system
- Malfunction in the coolant storage and treatment system
- Malfunction in the reactor boron water makeup system (RBWMS)

15.4.6.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.4.6, summarized here, in part, as follows:

The FSAR states that the positive reactivity insertion event resulting from the addition of nonborated water to the RCS from a failure in the reactor boron water makeup system or the chemical volume control system (CVCS) creates the most limiting events. In order to cover the entire range of operating modes for this event, three protection channels have been specifically defined to mitigate the consequences of the boron dilution event according to the reactor mode of operation. These safety-related protective channels are: Anti-dilution at power conditions, Modes 1 and 2; Anti-dilution in standard shutdown conditions, Modes 3, 4, and 5 with at least one RCP in operation; and Anti-dilution in shutdown conditions with RCPs secured, Modes 5 or 6 with the RCPs secured. These protection channels automatically isolate the CVCS, which terminates the boron dilution event, when the applicable protection system setpoint is reached.

The FSAR analyses determine the values of the PS setpoints for the three anti-dilution protection channels. Each of the three protection channels uses a different algorithm for defining the measured boron concentration. The control logic for this trip system is described in FSAR Tier 2, Section 7.3.1.2.11, “CVCS Isolation for Anti-Dilution.” The method for performing the analysis deterministically evaluates the uncertainty in each algorithm and applies it to the calculated critical boron concentration necessary to maintain shutdown margin. This method establishes the PS trip threshold in accordance with GDC 26. If the RCS boron concentration decreases below this threshold, indicating a possible dilution event or potential loss of shutdown margin, a PS signal is sent to isolate the CVCS, which terminates the boron dilution event.

The applicable acceptance criteria used in the FSAR analysis for this event are:

- Pressure in the reactor coolant and main steam systems is maintained below 110 percent of the design values.
- Fuel cladding integrity is maintained.
- The event does not generate a more serious plant condition without other faults occurring independently.

The applicant's FSAR analysis, performed for only the most limiting cases, concludes that the regulatory acceptance criteria are met, and there are no radiological consequences or releases from this event.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 15.4.6 are given in FSAR Tier 2, Chapter 16, Sections 3.1.1, "Shutdown Margin (SDM),"and 3.1.8, "Boron Dilution Protection (BDP)."

15.4.6.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.4.6, "Inadvertent Decrease in Boron Concentration in the Reactor Coolant System (PWR)," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.4.6.

1. GDC 10, as it relates to the reactor core and its coolant control and protection systems with appropriate design margin to assure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs.
2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety and of appropriate controls to maintain these variables and systems within prescribed operating ranges.
3. GDC 15, as it relates to the RCS and its auxiliary, control, and protection systems with sufficient design margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including AOOs.
4. GDC 26, as it relates to the capability of control rods to reliably control reactivity changes to assure that under conditions of normal operation, including AOOs, and with appropriate margin for malfunctions like stuck rods, SAFDLs are not exceeded.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.
2. Fuel cladding integrity must be maintained so the MDNBR remains above the 95/95 DNBR limit for pressurized-water reactors.

3. An incident of moderate frequency should not generate a more serious than moderate plant condition without other faults occurring independently.

4. If operator action is required to terminate the transient, the following minimum time intervals must be available between the time an alarm announces an unplanned moderator dilution and the time shutdown margin is lost:

   - During refueling: 30 minutes
   - During startup, cold shutdown, hot shutdown, hot standby, and power operation: 15 minutes

15.4.6.4 *Technical Evaluation*

FSAR Tier 2, Section 15.4 states that the malfunction of the reactor boron water makeup system leads to the worst combination of maximum dilution flow rate and minimum boron concentration. The injection of unborated water provides a slow injection of reactivity that is terminated by the insertion of control rods, injection of borated water, and isolation of the unborated water source. A maximum dilution rate of 0.1 ppm/sec corresponds to a maximum reactivity insertion rate of less than 0.1 pcm/sec. In comparison, the inadvertent rod withdrawal assumes a reactivity insertion rate of 2 pcm/sec. Therefore, the inadvertent dilution is bounded by the inadvertent rod withdrawal. The applicant does not provide a separate thermal-hydraulic coupled reactor kinetic analysis of the inadvertent dilution event. The staff finds this acceptable since the bounding analysis is identified. The inadvertent control rod withdrawal is described in FSAR Tier 2, Sections 15.4.1 and 15.4.2, and the results are acceptable.

The staff observes that the event is terminated by dedicated safety-related anti-dilution signals processed by the protective system. This trip is based on a real-time derivative of the homogeneous RCS boron concentration. The U.S. EPR uses automatic means to provide protection rather than manual actions typical of the operating fleet. Operator action is not required to terminate the event. The staff finds this acceptable.

FSAR Tier 2, Section 15.4 states that in Modes 1 or 2, control rods are available to mitigate the dilution. The following are available: DNB LCO alarm, DNBR RT, high core power level RT, LPD LCO alarm, high linear power density (HLPD) RT, High pressurizer level RT, and anti-dilution at power conditions trip (isolates CVCS).

FSAR Tier 2, Section 15.4 states that in Modes 3 or 4, dilution will continue until the anti-dilution protective system trip is reached. This trip is based on a real-time derivation of the homogeneous RCS boron concentration and is designed to actuate prior to a loss of shutdown margin. This trip automatically isolates the CVCS, which terminates the boron dilution event.

FSAR Tier 2, Section 15.4 states that in Modes 5 and 6, dilution will continue until the setpoint for the anti-dilution in shutdown conditions with RCPs secured trip is reached. This trip is based on the measured boron concentration being injected into the core. This trip automatically isolates the CVCS, which terminates the boron dilution event.

FSAR Tier 2, Section 15.4 states that the setpoints for actuation of the anti-dilution signal are biased conservatively, account for instrument and measurement uncertainties, and are acceptable. Analyses were performed assuming failure of one protective division as the single

failure. FSAR Tier 2, Section 15.4 explains how the setpoints are determined and how uncertainties are treated. The staff finds the method acceptable since it meets GDC 26.

15.4.6.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.4.6.6 *Conclusions*

The staff concludes that the design certification applicant has classified the CVCS malfunction event as a moderate-frequency event, which conforms to the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the analysis has used conservative input assumptions and methods.

The staff concludes that the analysis is bounded by the inadvertent control rod sub-bank withdrawal.

The staff concludes that the systems analysis conforms to RG 1.105 in that instrument uncertainties are included in the analytic limit, and complies with GDC 13 in that only safety-related trips are used in the analysis.

The staff concludes that the event evaluation considers single failures in accordance with RG 1.53.

The staff concludes that the event does not cause a more serious event and that sufficient safety-related equipment is available to shut down the reactor.

Therefore, the staff finds that:

1. GDC 10, as it relates to the reactor core and its coolant control and protection systems is met in that the analysis shows that SAFDLs are not challenged.

2. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems is met in that only safety-related equipment is taken credit for in the analysis.

3. GDC 15, as it relates to the RCS and its auxiliary, control, and protection systems is met in that the analysis shows that the design conditions of the RCPB are not exceeded.

4. GDC 26, as it relates to the capability of control rods to reliably control reactivity changes is met in that the analysis shows that in Modes 1 and 2 there is sufficient control rod worth to mitigate the boron dilution, with appropriate margin for malfunctions like stuck rods, exceeded.

15.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

15.4.7.1 *Introduction*

The inadvertent loading and operation of a fuel assembly in an improper location event is an AOO that is allowed to result in fuel failures provided that the radiological consequences are only a small fraction of the 10 CFR Part 100 criteria.

15.4.7.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.4.7, summarized here, in part, as follows:

FSAR Tier 2, Section 15.4.7 states that fuel pin enrichment or placement errors during fabrication are considered unlikely because of the use of strict procedural controls and verification processes. Fuel-loading errors into the reactor are also considered unlikely due to procedural controls and design features, which minimize the likelihood of the event. In addition, the in-core instrumentation systems can be used to potentially detect any errors in core loading. However, should an error be made and go undetected, it is possible in some reactor core loading designs for fuel rod failure limits to be exceeded. Therefore, fuel assembly loading error scenarios that cannot be detected by the in-core instrumentation are evaluated for radiological consequences.

The FSAR analysis concluded that less than two percent of the fuel pins would fail, and, therefore, the radiological consequences would only be a small fraction of the applicable 10 CFR Part 100 limits.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 15.4.7 are given in FSAR Tier 2, Chapter 16, Sections 3.2.2, “Nuclear Enthalpy Rise Hot Channel Factor”; 3.2.4, “Axial Offset (AO)”; 3.2.5, “Azimuthal Power Imbalance (API)”; 3.9.2, “Nuclear Instrumentation”; and 5.4, “Procedures.”

15.4.7.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.4.7, “Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position,” and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.4.7.

1. GDC 13, as it relates to providing instrumentation to monitor variables over anticipated ranges for normal operations, AOOs, and accident conditions.

2. 10 CFR Part 100, as it relates to offsite consequences resulting from reactor operations with an undetected misloaded fuel assembly.

15.4.7.4 *Technical Evaluation*

The inadvertent loading and operation of a fuel assembly in an improper location event is an AOO that is allowed to have fuel failures, provided several acceptance criteria are satisfied. This is in conformance with the SRP Regulatory Basis and is acceptable.

The acceptance criteria in NUREG-0800, Section 15.4.7 states that in the event the error is not detectable by the instrumentation system and fuel rod failure limits could be exceeded during normal operation, the offsite consequences should be a small fraction of the 10 CFR Part 100 criteria. A small fraction is interpreted to be less than 10 percent of the 10 CFR Part 100 reference values.

The applicant states:

> The aeroball measurement system (AMS) incore monitoring system provides protection against fuel loading errors by detecting power distribution anomalies. Administrative controls and the initial low-power flux map are the primary means used to determine if the core is loaded in conformance with the design. This is standard industrial practice and is acceptable contingent on approval of the aeroball system.

The aeroball system is discussed in AREVA TR ANP-10263P-A and evaluated in Section 4.4 of this report.

Analysis of the radial power distribution of the core with a postulated misloaded assembly is performed using approved methods described in Chapter 4, “Reactor,” using CASMO/SIMULATE.

The applicant states that the maximum number of assemblies with pins exceeding the nuclear enthalpy rise hot channel factor limit during an undetected misload is four. Four assemblies include conservatively up to 1,060 failed rods, or less than two percent of the pins in the core. Including all pins is highly conservative, because it is likely that only a small portion of each assembly actually violates the nuclear enthalpy rise hot channel factor limit.

The staff finds that given the intended checkerboard loading of fresh and burnt fuel, the limiting misload would result in fresh or reactive fuel next to other fresh or highly reactive fuel bundles. Therefore, an island of more reactive fuel is formed and the local power is greater than intended. The staff observes that the power distribution perturbation will promulgate across the core and will likely be detected. The staff finds that no more than all the fuel pins in four assemblies will experience DNBR is conservative given the typical power gradients across fuel assemblies and gradients within fuel assemblies. The staff concludes that the assumed damage of no more than four fuel assemblies is conservative based on the expected power distribution perturbation.

15.4.7.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.4.7.6 *Conclusions*

Suitable instrumentation and procedures are provided to detect a misloaded fuel assembly.

The misloading of a fuel assembly will not result in failure of more than two percent of the fuel pins in the core. Radiological consequences are acceptable.

Therefore, the U.S. EPR design complies with GDC 13, as it relates to providing instrumentation to monitor variables over anticipated ranges and 10 CFR Part 100, as it relates to offsite consequences resulting from reactor operations with an undetected misloaded fuel assembly.

15.4.8 Spectrum of Rod Ejection Accidents in a PWR

15.4.8.1 *Introduction*

The rod ejection accident is a postulated accident defined as the rupture of a control rod drive mechanism housing, resulting in the complete ejection of a rod control cluster assembly from the reactor core. The radiological consequences of the rod ejection accident are evaluated in FSAR Tier 2, Section 15.0.3.9, "Rod Ejection Accident."

15.4.8.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in FSAR Tier 2, Section 15.4.8, summarized here, in part, as follows:

The FSAR states that the consequences of the mechanical failure and ejection of a rod control cluster assembly are a rapid positive reactivity insertion, an increase in the local power peaking with high local energy deposition in the fuel assembly, and an initial pressure increase in the RCS. The power spike resulting from the RCCA ejection is quickly countered by the negative reactivity feedback when the fuel temperatures begin to increase. The event may lead to a RT, which will terminate the event.

For the spectrum of rod ejection accidents evaluated in the applicant's FSAR analyses, none resulted in fuel temperatures reaching either the limiting fuel melt temperature or the fuel enthalpy limits. For the events, that exceeded the DNBR limit, the number of fuel rod failures was less than the value allowed for the radiological release limit. The stresses due to the primary pressure response during the transients did not exceed the applicable ASME Code requirement.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 15.4.8 are given in FSAR Tier 2, Chapter 16, Sections 3.1.5 and 3.1.7.

15.4.8.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.4.8, "Spectrum of Rod Ejection

Accidents (PWR)," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Appendix B, Section 15.4.8 and Section 4.2.

1. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems over their anticipated ranges to assure adequate safety, and of appropriate controls to maintain these variables and systems within prescribed operating ranges.

2. GDC 28, as it relates to the effects of postulated reactivity accidents that result in neither damage to the RCPB greater than limited local yielding nor sufficient damage to impair significantly core cooling capacity.

3. 10 CFR 100.11, "Determination of exclusion area, low population zone, and population center distance," and 10 CFR 50.67, "Accident Source Term," establish radiation dose limits for individuals at the boundary of the exclusion area and at the outer boundary of the low population zone. The fission product inventory released from all failed fuel rods is an input to the radiological evaluation under NUREG-0800, Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors."

Consistent with NUREG-0800, Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," acceptance criteria adequate to meet the above requirements include:

1. Regulatory positions and specific guidelines necessary to meet the relevant requirements of GDC 28 are in RG 1.77 and SRP Section 4.2.

2. The maximum reactor pressure during any portion of the assumed excursion should be less than the value that result in stresses that exceed the "Service Limit C" as defined in the ASME B&PV Code.

15.4.8.4 *Technical Evaluation*

Methods

The applicant's approach for analyzing the spectrum of rod ejection accidents is discussed in ANP-10286P. This analysis method combines the results from neutronic, thermal-hydraulic, and plant simulations for a spectrum of initial plant conditions for the event. The methodology is used to determine the enthalpy definition in the fuel due to a rod efection accident, and is based on the 3-D spatial kinetics methodology.

For rod ejection overpressurization analyses, the S-RELAP5 computer code is used to determine the peak pressure response of the primary system to the RCCA ejection event. The methodology used to analyze the overpressurization event is based on the point kinetics model contained in S-RELAP5. Point kinetics conservatively predicts the reactor dynamics relative to a more detailed three dimensional space-time model. Therefore, the peak pressure calculation uses an accepted code and conservative input parameters and is acceptable. The methodology is separate from the methodology used in the other rod ejection accident analysis cases, which are based on the 3-D spatial kinetics methodology presented in AREVA TR ANP-10286P. Review of this topical report is currently ongoing. **RAI 344, Question 04.03-28, which is associated with the ongoing review of TR ANP-10286P, is being tracked as an open item**.

The S-RELAP5 analysis described in FSAR Tier 2, Section 15.4, demonstrates that the primary pressure does not exceed 120 percent of the system design pressure of 21.072 MPa (3,056 psia). The staff finds that the analyses were performed at HFP, 60 percent power, and HZP with a conservative positive MTC and a conservatively biased positive Doppler coefficient. The staff concludes that conservative modeling and input parameters are used in the analysis of the peak RCS pressure, and the results are acceptable.

FSAR Tier 2, Section 15.4 states that:

> The power spike resulting from the reactivity insertion associated with the rod ejection is quickly countered by Doppler reactivity feedback, when the fuel temperatures begin to increase, and is terminated by an RT. The probable trips are the safety-related high positive neutron flux rate, low neutron flux doubling time (intermediate range), or high neutron flux (intermediate range) signals. Although the initial increase in power occurs too fast to be impacted by the RT, the RT terminates the event and limits the damage from the pulse.

The staff notes that this is the classic explanation of the dynamics of the ejected rod accident.

Analyses of the enthalpy deposition in the fuel, cladding temperature, and enthalpy subchannel power used to compute DNBR were performed for a range of ejected rod worth, and initial reactor power at beginning of life (BOL) and end of life (EOL). The enthalpy deposition is predicted using AREVA 3-D space-time kinetics. The staff notes that the results are typical of those obtained using this method and are acceptable. The staff finds that the analyses results meet the PWR pellet/cladding mechanical interaction (PCMI) Fuel Cladding Failure Criteria shown in NUREG-0800, Section 4.2, Appendix B.

In order to better understand the accident analysis, in RAI 29, Question 15.04.08-1, the staff requested that the applicant confirm that a variety of control rod locations were considered. It appeared that the ejected rod accident has been performed assuming the center control rod is ejected.

In an October 1, 2008, response to RAI 29, Question 15.04.08-1, the applicant stated that the center control rod is not the only rod considered for the rod ejection accident. The neutronics model is a full core nodal model. Each of the inserted control rods is ejected to find the maximum rod worth locations. The staff finds that this method is comprehensive and, therefore, acceptable.

The thermal-hydraulic code LYNXT is used to calculate the thermal effects of the ejected rod transient. Power peaking uncertainty and allowance factors are applied to the fuel rod powers supplied to LYNXT. Additional factors, such as hot channel and flow misdistribution factors, are applied in the LYNXT model input. This response is acceptable, since it utilizes accepted codes and industry wide methods. Therefore, the staff considers RAI 29, Question 15.04.08-1 resolved.

In RAI 29, Question 15.04.08-2, the staff requested that the applicant provide an explanation of the differences in the ejected rod worths shown in FSAR Tier 2, Table 15.4-15, “Rod Ejection Accident Overpressurization Analysis – Key Input Parameters,” and those shown in FSAR Tier 2, Tables 15.4-17, “Rod Ejection Accident DNBR Analysis - Ejected Rod Analysis Results for BOC,” and 15.4-18, “Rod Ejection Accident DNBR Analysis - Ejected Rod Analysis Results

for EOC." In a September 2, 2008, response to RAI 29, Question 15.04.08-2, the applicant stated:

> FSAR Tier 2, Table 15.4-15 includes inputs for the plant system overpressurization criterion, while the Tables 15.4-17 and 15.4-18 include inputs and results regarding the DNBR and enthalpy rise criteria. The overpressurization analysis uses different codes and methods than the analysis used to calculate enthalpy rise and DNBR performance. The overpressure analysis uses S-RELAP5, while the enthalpy is computed with the methodology in ANP-10286P. In the rod ejection analyses, the delayed neutron fraction and fuel Doppler coefficient are adjusted to have lower magnitudes in order to enhance the period and peak value of the power pulse.

The response shows conservative selection of input values and is acceptable. Therefore, the staff considers RAI 29, Question 15.04.08-2 resolved.

In order to confirm conservative acceptance criteria, in RAI 29, Question 15.04.08-3, the staff requested that the applicant verify that the 110 cal/gm criterion shown in FSAR Tier 2, Table 15.4-14, "Rod Ejection Accident DNBR Analysis – Ejected Rod Analysis Limits for U.S. EPR" is the PCMI limit of ANP-10286P. In an October 1, 2008, response to RAI 29, Question 15.04.08-3, the applicant stated:

> ANP-10286P, "U.S. EPR Rod Ejection Accident Methodology Topical Report," Section 2.1.1 addresses the enthalpy (cal/g) limit for M5™ clad fuel. The maximum corrosion for M5™ cladding at end of life burnups is 0.035 mm (0.001378 in.). The corresponding oxide-to-wall thickness ratio is 0.061. Using the pellet/cladding mechanical interaction fuel cladding failure criteria of Standard Review Plan, Section 4.2, the PCMI failure limit is 110 cal/g. A single cal/g limit for all fuel is used to keep the analysis independent of burnup by taking the maximum end of life burnup corrosion.

The response meets the regulatory guidance and is acceptable. Therefore, the staff considers RAI 29, Question 15.04.08-3 resolved.

In RAI 29, Question 15.04.08-4, the staff requested that the applicant clarify the legibility of selected FSAR Tier 2, Section 15.4 figures. In a September 2, 2008, response to RAI 29, Question 15.04.06-1, the applicant stated that additional figures would be added for clarity. The figures are provided in Revision 1 of the FSAR and, therefore, the staff considers RAI 29, Question 15.04.08-4 resolved.

To better understand the ejected rod analysis, in RAI 29, Question 15.04.08-5, the staff requested that the applicant identify the computer codes used in the analysis. In a September 2, 2008, response to RAI 29, Question 15.04.08-5, the applicant stated that the computer codes used are described in AREVA TR ANP-10263P-A. Each code has supporting topical reports approved for use in PWR applications. The computer codes used for the rod ejection analysis were S-RELAP5, COPERNIC, LYNXT, and NEMO-K. The application of these codes, as well as modifications of the codes, is described in AREVA TR ANP-10286P, for the rod ejection accident analysis. The applicant also provided a description of the nodalization. This response is acceptable, since approved codes and methods are used. Therefore, the staff considers RAI 29, Question 15.04.08-5 resolved. The staff notes that the review of

TR ANP-10286P is currently ongoing. **RAI 344, Question 04.03-28, which is associated with the ongoing review of TR ANP-10286P, is being tracked as an open item**.

In RAI 29, Question 15.04.08-6, the staff requested that the applicant identify the core location of the limiting control rod in both the initial and equilibrium core. In an October 1, 2008, response to RAI 29, Question 15.04.08-6, the applicant stated that the rod cluster control assembly in the N05 fuel location is the limiting rod ejection location for the equilibrium core design. All the cases in the reference NEMO-K analysis assume the rod is ejected from this location. The analysis parameters are set to bound all cycles. Furthermore, the results presented in AREVA TR ANP-10286P are shown not to be sensitive to the ejected rod core location when the other key parameters remain the same. This response is acceptable. Therefore, the staff considers RAI 29, Question 15.04.08-6 resolved.

In order to confirm that the ejected rod analysis uses conservative input, in RAI 29, Question 15.04.08-7, the staff requested that the applicant provide the physical basis to show that the least negative Doppler temperature coefficient (DTC) has been used in the calculations. In a September 2, 2008, response to RAI 29, Question 15.04.08-7, the applicant stated that:

> The DTCs are calculated from the NEMO-K models used for each of the ejected rod transient simulations. The NEMO-K model used for the safety analysis simulations used a conservative fuel temperature response of the cross sections to bound the uncertainties on the DTC and to bound cycle-to-cycle variations.
>
> To verify that a particular cycle design is bounded by the ejected rod analysis, the unmodified cycle specific values for the DTC are reduced by the uncertainty and compared to the analyzed values at BOC, EOC, HFP, and HZP. This means the DTC for the designed cycle will be larger in magnitude by the uncertainty than the value analyzed. An example is shown in AREVA TR ANP-10286P.

This response is acceptable since it describes a technically sound and conservative methodology. Therefore, the staff considers RAI 29, Question 15.04.08-7 resolved.

In order to better understand the analysis, in RAI 29, Question 15.04.08-8, the staff requested that the applicant identify the S-RELAP5 model used to determine the peak pressure response of the primary system to the RCCA ejection event. In a September 2, 2008, response to RAI 29, Question 15.04.08-8, the applicant stated that the non-LOCA S-RELAP5 model and nodalization described in AREVA TR ANP-10263P-A was used. This response is acceptable, since approved methods were used. The staff observes that the pressure response is insensitive to the details of the model. Therefore, the staff considers RAI 29, Question 15.04.08-8 resolved.

In RAI 29, Question 15.04.08-9, the staff requested that the applicant provide a description of the xenon condition at the start of the rod ejection accident analysis and a justification that the assumed xenon condition conservative. In a September 2, 2008, response to RAI 29, Question 15.04.08-9, the applicant stated that the HFP xenon concentration is skewed, so that the power offset represents the maximum offset achievable at HFP. The staff observes that the top peaked shape provides conservative input for the analysis because it results in maximizing the ejected rod worth and Fq, and therefore minimizing DNBR. Therefore, the staff considers RAI 29, Question 15.04.08-9 resolved.

15.4.8.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.4.8.6 *Conclusions*

The staff concludes that the applicant has classified the ejected control rod as a postulated accident in conformance with the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the analysis has used conservative input assumptions and methods to maximize the fuel rod enthalpy deposition and peak reactor system pressure.

The staff finds that the analysis shows that the enthalpy rise is less than the PCMI failure limit in NUREG-0800, Section 4.2, Appendix B.

The staff concludes that the analysis shows that peak primary pressure does not exceed 120 percent of the system design pressure.

The staff concludes that the event does not cause a more serious event. Sufficient safety-related equipment is available to shut down the reactor.

The staff concludes that the analyses were performed with reviewed and approved codes and methods.

Therefore, with the exception of the open items identified above, the staff finds that:

1. GDC 13, as it relates to the availability of instrumentation to monitor variables and systems is met in that only safety-related instrumentation is assumed available in the safety analysis.

2. GDC 28, as it relates to the effects of postulated reactivity accidents that result in neither damage to the RCPB greater than limited local yielding nor sufficient damage to impair significantly core cooling capacity is met in that the peak pressure predicted by the analysis does not exceed 120 percent of the system design pressure.

3. 10 CFR 100.11, “Determination of exclusion area, low population zone, and population center distance,” and 10 CFR 50.67, “Accident Source Term,” is discussed in Section 15.0.3 of this report.

15.5 Increase in Reactor Coolant Inventory

FSAR Tier 2, Section 15.5 subsections describe the following AOOs that increase RCS inventory during power operation:

- FSAR Tier 2, Section 15.5.1, “Inadvertent Operation of ECCS or EBS”

- FSAR Tier 2, Section 15.5.2, ”Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory”

Transient and accident categorization and classification are discussed in FSAR Tier 2, Section 15.0.0. The computer codes used to analyze DBAs are discussed in FSAR Tier 2,

Section 15.0.2. Radiological consequences for DBAs are discussed in FSAR Tier 2, Section 15.0.3.

15.5.1 Inadvertent Operation of Emergency Core Cooling System, Extra Borating System, or Chemical and Volume Control System

This section documents the staff's review of FSAR Tier 2, Sections 15.5.1 and 15.5.2.

15.5.1.1 *Introduction*

The inadvertent operation of the ECCS resulting from the spurious actuation of the safety injection system (SIS) or the inadvertent actuation of the EBS or the CVCS are classified as AOOs. Operation of the EBS or the CVCS leads to an increase in the inventory of the RCS.

15.5.1.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.5.1, summarized here, in part, as follows:

FSAR Tier 2, Section 15.5 states that the spurious or inadvertent actuation of the SIS at pressure levels above the shutoff heads of the ECCS pumps and the accumulator injection setpoints would not result in an increase in RCS inventory. A spurious safety injection (SI) signal causes an RT, initiates partial cooldown, and isolates the RCS boundary. The FSAR states that this scenario is bounded by the overcooling analyses presented in FSAR Tier 2, Section 15.1.

FSAR states that the U.S. EPR has a manually actuated safety-related extra-borating system (EBS) that if inadvertently operated could result in an increase in RCS inventory. The EBS consists of two divisions, each with a high-pressure, positive displacement pump and an EBS tank. During normal operation, the pumps are in standby mode and must be started manually. However, assuming the pumps were started and the isolation valves opened, RCS inventory would increase. Under this assumption, the pressurizer level increases until an RT protection system signal is received. The resulting increase in pressure causes the PSRVs to open. This controls primary system pressure. After 30 minutes, operator action is assumed to be taken in order to terminate the event and place the plant in a controlled state by either de-energizing the EBS pumps or closing the EBS isolation valves. No other operator actions are credited.

RCS inventory and pressure and pressurizer water level are controlled during normal operation by the CVCS. The CVCS utilizes charging pumps and the letdown control valve to control the RCS inventory. The pressurizer level control system continuously adjusts the CVCS letdown flow while one of the two charging pumps provides makeup. The second charging pump is used whenever the PZR level is sufficiently low, and the letdown line is isolated if the PZR level falls below the letdown isolation setpoint. The principal assumption for analyzing this event, in order to maximize the increase in RCS inventory, is that both charging pumps are operating and the letdown line is isolated.

The analysis contained in FSAR Tier 2, Section 15.5 concludes that the regulatory acceptance criteria are met, and there are no radiological consequences or releases from this event.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: The Technical Specifications associated with FSAR Tier 2, Section 15.5.5 are given in FSAR Tier 2, Chapter 16, Sections 3.5.1, "Accumulators"; 3.5.2, "ECCS – Operating"; and 3.5.4, "In-Containment Refueling Water Storage Tank − Operating."

15.5.1.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Sections 15.5.1-15.5.2, "Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Sections 15.1-15.2.

1. GDC 10, which requires that the reactor core and associated coolant control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of AOOs.

2. GDC 13, which requires, in part, that the effect of instrumentation shall be provided to monitor variables and systems over their anticipated ranges for AOOs to assure adequate safety. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, which requires that the RCS and its associated auxiliary control and protection systems be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operations, including AOOs.

4. GDC 26, which requires, in part, the reliable control of reactivity changes to assure that SAFDLs are not exceeded under conditions of normal operation, including AOOs, with appropriate margin formal functions, such as stuck rods.

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values in accordance with the ASME B&PV Code.

2. Fuel cladding integrity should be maintained by ensuring that the MDNBR remains above the 95/95 DNBR limit for PWRs.

3. An AOO should not generate a more serious plant condition without other faults occurring independently.

15.5.1.4 *Technical Evaluation*

The analysis of the inadvertent operation of the SIS, or inadvertent operation of the EBS or CVCS event uses the approved non-LOCA analytical methodology described in AREVA TR ANP-10263P-A and the S-RELAP5 computer code to simulate the event.

The inadvertent operation of the EBS or CVCS results in a slow increase in the RCS inventory and an increase in pressurizer levels. FSAR Tier 2, Section 15.5 states that the RTs on the

safety-related high pressurizer level signal after 13 minutes following CVCS actuation and more than 20 minutes following EBS actuation. Isolation of the CVCS is automatic while credit is taken for operator action after one-half hour to secure the EBS. The analysis contained in FSAR Tier 2, Section 15.5 assumes a TT follows an RT and that LOOP occurs at TT. The staff notes that the LOOP causes coastdown of the RCPs minimizing coolant loop flow and heat transfer to the SGs and maximizes primary system pressure.

The staff observes that the LOOP is not taken as the independent single failure in conformance with the guidance in RG 1.206 and NUREG-0800. A single failure is assumed, specifically failure of an EFW division. The non-safety-related PZR heaters and sprays are assumed to operate as designed, since these result in a more limiting transient. The analysis uses conservative input and assumptions about the availability of equipment and is acceptable.

The FSAR Tier 2, Table 15.5-3, "Inadvertent Operation of the EBS-Sequence of Events," shows a maximum PZR level reached (98.7 percent) at 3,422 seconds, and FSAR Tier 2, Table 15.5-6, "CVCS Malfunction that Increases RCS Inventory-Sequence of Events," shows a maximum PZR level reached (98.6 percent) at 2,142 seconds. The analysis assumes the operator terminated the event after 30 minutes by de-energizing EBS pumps or closing the isolation valves. The staff notes that the primary system pressurization and PZR level increase are due to a mismatch of decay heat and heat removal from the secondary side. Eventually, the emergency feedwater system refills the SGs, increasing heat transfer to the SGs, and terminating the increase in pressurizer level. Pressurizer pressure is limited by the PSRVs. The staff finds that, because of the long time for operator action, it can be concluded that the PZR will not go water solid.

In RAI 29, Question 15.05.01-15.05.02-1, the staff requested that the applicant explain the basis for the EBS and CVCS flow used in the accident analysis. In a September 2, 2008, response to RAI 29, Question 15.05.01-15.05.02-1, the applicant stated that the maximum flow was used in the analysis, and provided additional information about system lineup and valve position. In RAI 29, Question 15.05.01-15.05.02-2, the staff requested that the applicant explain why there was unequal CVCS flow to the RCS loops. In a September 2, 2008, response to RAI 29, Question 15.05.01-15.05.02-2, the applicant stated that the flow rates are not the same due to the difference in losses through the two loops based on actual line-loss calculations, based on the pipe lengths. The response is acceptable since it explains the difference in flow. Therefore, the staff considers RAI 29, Questions 15.05.01-15.05.02-1 and 15.05.01-15.05.02-2 resolved.

15.5.1.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.5.1.6 *Conclusions*

The staff concludes that the applicant has classified the inadvertent operation of the SIS or inadvertent operation of the EBS or CVCS as moderate-frequency events, which conforms to the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the inadvertent operation of the safety injection system will not cause a transient, since the discharge pressure of the SIS is less than the RCS operating pressure.

The staff concludes that, with respect to inadvertent operation of the EBS or CVCS:

- The analysis has used conservative input assumptions and methods to minimize calculated DNBR and maximize primary pressure. SAFDLs are not challenged. Overpressure of the primary system is limited by the PSRVs to acceptable values.
- The most limiting single failure has been assumed in the analysis in conformance with RG 1.53. LOOP has been taken as a consequential failure rather than a single failure which conforms to the guidance in RG 1.206 and NUREG-0800
- The systems analysis conforms to RG 1.105 in that instrument uncertainties are included in the analytic limit, and complies with GDC 13 in that only safety-related trips are used in the analysis.
- The event does not cause a more serious event, and that sufficient safety-related equipment is available to shut down the reactor.
- The analyses were performed with reviewed and approved codes and methods.

Therefore, with exception of the open items mentioned above, the staff finds that:

1. GDC 10, which requires that the reactor core and associated coolant control, and protection systems be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded is met in that the analysis shows that DNBR limits are never challenged during these events.
2. GDC 13, which requires, in part, that the effect of instrumentation shall be provided to monitor variables and systems, is met in that the analysis assumes the availability of only safety-related instrumentation.
3. GDC 15, which requires that the RCS and its associated auxiliary control and protection systems be designed with sufficient margin, is met in that the analysis shows that the design pressure of the RCPB is not exceeded.
4. GDC 26, which requires, in part, the reliable control of reactivity changes to assure that SAFDLs are not exceeded is met in that the analysis assumes margin for failures such as the most reactive stuck rod.

15.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

Review of this section of the FSAR is documented under Section 15.5.1 of this report.

15.6 Decrease in Reactor Coolant Inventory Events

Several AOOs and postulated accident events cause a decrease in reactor coolant inventory and are included in the following FSAR Tier 2, Section 15.6, “Decrease in Reactor Coolant Inventory,” subsections:

- FSAR Tier 2, Section 15.6.1, “Inadvertent Opening of a Pressurizer Safety Relief Valve”
- FSAR Tier 2, Section 15.6.2, “Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment”
- FSAR Tier 2, Section 15.6.3, “Steam Generator Tube Failure (PWR)”
- FSAR Tier 2, Section 15.6.4, “Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)” (Not applicable to the U.S. EPR)
- FSAR Tier 2, Section 15.6.5, “Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary”

Transients and accidents are discussed in FSAR Tier 2, Section 15.0.0. The computer codes used to analyze DBAs are discussed in FSAR Tier 2, Section 15.0.2. Radiological consequences for DBAs are discussed in FSAR Tier 2, Section 15.0.3.

15.6.1 Inadvertent Opening of a Pressurizer Safety Relief Valve

15.6.1.1 *Introduction*

The inadvertent opening of a pressurizer safety valve (IOPSRV) event is defined as the spurious opening of a normally closed pressurizer safety and relief valve. The IOPSRV event is considered an AOO.

15.6.1.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.6.1, summarized here, in part, as follows:

The FSAR states that the PSRVs serve as both relief and safety valves, and as a result, no isolation valves are provided to isolate the pressurizer relief line from the pressurizer. Therefore, an IOPSRV event results in a loss of reactor coolant inventory that cannot be isolated or offset by the CVCS. The design of the PSRVs during power operation is such that a single failure can affect only one PSRV. An IOPSRV event causes primary system depressurization and a decrease in reactor coolant inventory eventually resulting in an RT and safety injection. Following safety injection, the reactor coolant inventory is increased and a controlled cooldown and depressurization can be initiated by the operator.

The FSAR analysis concludes that the regulatory acceptance criteria are met, and there are no radiological consequences or releases from this event.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: There are no Technical Specifications for this area of review.

15.6.1.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.6.1, "Inadvertent Opening of a PWR Pressurizer Pressure Relief Valve or a BWR Pressure Relief Valve," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.6.1.

1. GDC 10, as it relates to designing the RCS with appropriate margin to assure that SAFDLs are not exceeded during normal operations, including AOOs.

2. GDC 13, as it relates to providing instrumentation to monitor variables and systems over their anticipated ranges for normal operation to assure adequate safety, and to providing appropriate controls to maintain these variables and systems within prescribed operating ranges.

3. GDC 15, as it relates to designing the RCS and associated auxiliary systems with sufficient margin to assure that the design conditions of the RCPB are not exceeded during normal operations, including AOOs.

4. GDC 26, as it relates to providing a reactivity control system capable of reliably controlling reactivity changes during manual operations and AOOs so the specified fuel design limits are not exceeded.

5. 10 CFR 52.47(a), "Contents of applications; technical information," as it relates to demonstrating compliance with any technically relevant portions of the Three Mile Island-related requirements set forth in 10 CFR 50.34(f)(1)(vi), and 10 CFR 50.34(f)(1)(iii).

Acceptance criteria adequate to meet the above requirements include:

1. Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of the design values.

2. Fuel cladding integrity is maintained if the MDNBR remains above the 95/95 DNBR limit.

3. An AOO should not develop into a more serious plant condition without other faults occurring independently.

15.6.1.4 *Technical Evaluation*

The IOPSRV event uses the approved analytical methodology described in AREVA TR ANP-10263P-A and the S-RELAP5 computer code to simulate the event. The low DNB channel algorithm is simulated to predict RT and the adequacy of the dynamic compensation of the algorithm in conformance with AREVA TR ANP-10287P.

The staff confirmed that the analysis is performed with conservative input assumptions related to SG tube plugging (five percent), reactor kinetics, and control rod scram worth. Following an RT and a TT, it is conservatively assumed that offsite power is lost and that the RCPs coastdown which minimizes heat removal from the primary system across the SGs. This conforms to the guidance in RG 1.206 and NUREG-0800. The applicant has taken the worst

single failure as the failure of one emergency diesel generator (EDG) resulting in the unavailability of one division of SIS (medium head safety injection (MHSI), low head safety injection (LHSI), and emergency feedwater system (EFWS)). Another division of SIS is assumed out of service for maintenance. Operator actions are credited at 30 minutes into the event to align EFWS flow from the two operational divisions of EFWS to the four SGs. Later, operator actions are necessary to transition the plant from a controlled state to a safe-shutdown condition.

In RAI 34, Question 15.06.01-1, the staff requested that the applicant provide the DNBR as a function of time for all FSAR Tier 2, Section 15.6 analyses for which it is a key parameter. In a September 24, 2008, response to RAI 34, Question 15.06.01-1, the applicant revised the FSAR to include Figure 15.6-93, "IOPSRV – Representative Plot of Minimum Normalized Minimum DNBR and Maximum LPD Normalized to the SAFDL," for the inadvertent opening of a PSRV event. In a September 24, 2008, response to RAI 34, Question 15.06.01-1, the applicant stated that the FSAR Tier 2, Section 15.6.1.2, "Method of Analysis and Assumptions," and FSAR Tier 2, Section 15.6.1.3, "Results," will be revised to clarify the DNB and LPD trip functions. Revision 1 of the U.S. EPR FSAR shows the changes, figures and technical explanation. The applicant's response is acceptable since the requested information is provided. Therefore, the staff considers RAI 34, Question 15.06.01-1 resolved.

In RAI 34, Question 15.06.01-2, the staff requested that the applicant provide an assessment showing why the parameters, initial condition, and single failures assumed for each event are those which lead to the most conservative results. In a September 24, 2008, response to RAI 34, Question 15.06.01-2, the applicant stated that this RAI is addressed in the response to RAI 34, Questions 15-1, 15.-2, 15-3, 15-4, 15-5, 15-6, and 15-7. The responses to these RAIs are discussed in Section 15.0 of this report and are found by the staff to be acceptable. Therefore, the staff considers RAI 34, Question 15.06.01-2 resolved.

In RAI 34, Question 15.06.01-3, the staff noted that the event presented in FSAR Tier 2, Section 15.6.1 shows an initial power increase due to boron reactivity feedback and requested an explanation of how the boron feedback as well as axial and radial power profiles are treated conservatively. In a September 24, 2008, response to RAI 34, Question 15.06.01-3, the applicant stated that its analysis of the IOPSRV event used nominal axial and radial power profiles, in accordance with its approved non-LOCA methodology. The applicant also presented an additional analysis of the IOPSRV event using the SBLOCA methodology and concluded that the results of the analysis were unchanged. The applicant stated:

> The boron feedback effect is modeled in the analysis via the relationship between the moderator temperature and density. Analyses are performed with the moderator temperature coefficient as the most positive at BOC or as the most negative at EOC to determine the conservative boron feedback effect.
>
> The inadvertent opening of a PSRV analysis presented in the FSAR Tier 2, Section 15.6.1 uses nominal axial and radial power profiles without additional conservatism. However, a sensitivity study was conducted utilizing the SBLOCA methodology to evaluate the IOPSRV event with added conservatism. The core model in the SBLOCA method consists of a three-region radial representation of the core and considers a top-peaked axial power profile. In addition, the Moody critical flow model and a 20 percent multiplier on the decay heat are other conservative assumptions of the SBLOCA method. Conservative treatment of

the break flow through the PSRV and the reactor power decay heat is already considered in the reference analysis.

The results from the IOPSRV event analysis using the SBLOCA methodology predict no clad heat-up and calculated a hot channel core exit void fraction of about 0.6. The IOPSRV event analysis with the non-loss of coolant analysis methodology and with conservative PSRV flow and core decay heat assumptions produces results comparable to those obtained by using the SBLOCA methodology. Therefore, the penalizing assumptions of the SBLOCA analysis method do not significantly affect the response of the IOPSRV event.

The staff determines the applicant's response, to be acceptable since the principle dynamic of a SBLOCA is the break size and location, and these parameters are fixed in the IOPSRV and it would be expected that other sensitivities would be small as is shown in the applicant's calculations. Therefore, the staff considers RAI 34, Question 15.06.01-3 resolved.

15.6.1.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.6.1.6 *Conclusions*

The staff concludes that the applicant has identified the inadvertent opening of the pressurizer safety valve as a moderate-frequency event, which conforms to the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the most limiting single failure has been assumed in the analysis in conformance with RG 1.53. LOOP has been taken as a consequential failure rather than a single failure which conforms to the guidance in RG 1.206 and NUREG-0800.

The staff concludes that the systems analysis conforms to RG 1.105 in that instrument uncertainties are included in the analytic limit, and complies with GDC 13 in that only safety-related trips are used in the analysis.

The staff concludes that the analysis has used input assumptions and methods to minimize primary system inventory, minimize calculated DNBR, and maximize primary and secondary system pressure. The analysis uses approved methodologies and computer codes.

The staff concludes that the analysis shows that the pressure in the reactor coolant and main steam systems is maintained below 110 percent of the design values and that fuel cladding integrity is maintained, since the MDNBR remains above the 95/95 DNBR limit.

The staff concludes that the event does not cause a more serious event. Sufficient safety-related equipment is available to shut down the reactor.

Therefore, the staff finds that:

1. GDC 10, as it relates to designing the RCS with appropriate margin to assure that SAFDLs are not exceeded is met in that the analysis shows that LPD and DNBR criteria are not exceeded.

2. GDC 13, as it relates to providing instrumentation to monitor variables and systems over their anticipated ranges is met in that the safety analysis only assumes the availability of safety-related instrumentation.

3. GDC 15, as it relates to designing the RCS and associated auxiliary systems with sufficient margin to assure that the design conditions of the RCPB are not exceeded during normal operations in that the analysis shows that neither the pressure not cooldown rate of the RCPB challenge acceptance criteria.

4. GDC 26, as it relates to providing a reactivity control system capable of reliably controlling reactivity changes is met in that the analysis takes credit for the RPS with the most reactive rod stuck out.

5. 10 CFR 52.47(a), "Contents of applications; technical information," as it relates to demonstrating compliance with any technically relevant portions of the Three Mile Island-related requirements set forth in 10 CFR 50.34(f)(1)(vi), and 10 CFR 50.34(f)(1)(iii) is met in that RCP pump seal cooling is not interrupted during the event, and that the ratio of pressurizer volume to RCS volume is greater in the U.S. EPR than in the operating fleet and this greater volume reduces the likelihood of an AOO leading to a IOPSRV.

15.6.2 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

15.6.2.1 *Introduction*

The rupture of small lines carrying primary coolant outside containment is analyzed as non-mechanistically postulated accidents that are evaluated for radiological consequences.

15.6.2.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.6.2, summarized here, in part, as follows:

The FSAR states that the rupture of small lines analyzed for this postulated accident are limited to the CVCS and the nuclear sampling system lines. Breaks of other size lines are less limiting or do not transport primary reactor coolant. The U.S. EPR design has no instrument lines that carry primary coolant outside of containment. No operator action is credited for the first 30 minutes of the postulated accident.

The radiological consequences of these events are addressed in FSAR Tier 2, Section 15.0.3.5, "Small Line Carrying Primary Coolant Break Outside of the Reactor Building Accident."

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: There are no Technical Specifications for this area of review.

15.6.2.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.6.2.

1. GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment"

2. 10 CFR Part 100, related to the radiological consequences of a small line break carrying primary coolant outside containment

Acceptance criteria adequate to meet the above requirements include:

- RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment," which defines isolation capability outside containment and other requirements for those lines that are exempt from GDC 55 instrument line penetrations.

15.6.2.4 *Technical Evaluation*

A non-mechanistic radiological analysis is presented in FSAR Tier 2, Section 15.0.3.

Containment isolation is addressed in FSAR Tier 2, Section 6.2.4, "Containment Isolation Systems."

The U.S. EPR design does not have instrument lines penetrating containment and, therefore, meets RG 1.11.

15.6.2.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.6.2.6 *Conclusions*

The relevant conclusions are contained in Section 15.0.3.6 of this report.

15.6.3 Steam Generator Tube Failure (PWR)

15.6.3.1 *Introduction*

The SGTR is a postulated accident defined as the double-ended rupture of a single SG tube. The principal acceptance criterion for this event is to maintain the radiological releases below acceptable limits provided in 10 CFR Part 100. A secondary criterion is to prevent overfill of the SG secondary in order to prevent water from entering the steamlines.

15.6.3.2 *Summary of Application*

FSAR Tier 1: There are no FSAR Tier 1 entries for this area of review.

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.6.3, summarized here, in part, as follows:

The FSAR states that the SGTR postulated accident is assumed to occur in the shortest SG tube, near the tube sheet location, in order to maximize break flow. After the STGR, primary coolant from the RCS begins to enter the secondary system through the ruptured SG tube driven by the large pressure differential between the RCS and the secondary side of the SG. This results in an increase in the affected SG level (inventory), pressure, and radiological content. The loss of inventory from the RCS begins to depressurize the RCS and decrease the pressurizer level. The chemical volume and control system charging pumps inject more water into the cold legs of the RCS in order to maintain PZR level. If one CVCS charging pump cannot keep up with the break flow and the PZR level continues to decrease, a second charging pump (normally on standby) is automatically started on low-PZR level. The CVCS letdown flow is automatically reduced to its minimum value in response to the decreasing PZR level. The CVCS pumps are automatically switched to the in-containment refueling water storage tank on low level in the volume control tank. If the combined CVCS charging pumps are able to offset the RCS coolant loss through the single tube rupture, then the operator is assumed to trip the reactor before the RCS pressure decreases sufficiently to reach the automatic RT setpoint. If the CVCS charging pumps are not available or adequate, an automatic RT will be initiated on one of several protection system setpoints for RT, and the PZR heaters are de-energized as PZR level continues to decrease.

The FSAR analyzes several scenarios in order to ensure that the most limiting conditions are considered. However, for any of the initiating scenarios assumed, the termination of the SGTR event requires minimizing the pressure differential between the RCS and the ruptured SG tube, so that radiological releases can be maintained below acceptable limits and the affected SG does not overfill. The EBS is placed into operation in order to provide adequate negative reactivity to prevent return to criticality from the cooldown of the RCS. Cooldown and depressurization of the RCS continue until the plant can be placed on the RHR system and taken to a shutdown condition.

The applicant's FSAR analysis concludes that even with conservative initial equipment and plant conditions, the SGTR event can be controlled through a combination of automatic and operator actions with radiological releases being below 10 CFR Part 100 regulatory limits (or within the limits of 10 CFR 50.67 for Alternate Source Term) and that the affected SG liquid level increase does not lead to more severe consequences.

ITAAC: There are no ITAAC items for this area of review.

Technical Specifications: There are no Technical Specifications for this area of review.

15.6.3.3 *Regulatory Basis*

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.6.3.

- 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance."

Acceptance criteria adequate to meet the above requirements extracted from SRP Section 15.0.3 include:

1. 10 CFR Part 50, Section 50.34(a)(1), “Contents of applications; technical information,” as it relates to the evaluation and analysis of the offsite radiological consequences of postulated accidents with fission product release.

2. GDC 19, 10 CFR Part 50, Appendix A, “Control room,” as it relates to maintaining the control room in a safe condition under accident conditions by providing adequate protection against radiation.

3. 10 CFR Part 100, Section 100.21, “Non-seismic siting criteria,” as it relates to the evaluation and analysis of the radiological consequences of postulated accidents for the type of facility to be located at the site in support of evaluating the site atmospheric dispersion characteristics.

4. 10 CFR Part 50, Appendix E, Paragraph IV.E.8, “Emergency Planning and Preparedness for Production and Utilization Facilities,” as it relates to adequate provisions for an onsite technical support center (TSC) from which effective direction can be given and effective control can be exercised during an emergency.

15.6.3.4 *Technical Evaluation*

The STGR event uses the approved analytical methodology described in AREVA TR ANP-10263P-A and the S RELAP5 computer code to calculate the transient thermal and hydraulic response of the primary and secondary systems. The methodology and results of the radiological analysis are presented in FSAR Tier 2, Section 15.0.3.

FSAR Tier 2, Section 15.6.3 states that the analysis for the SGTR event consists of two parts, a SG overfill calculation and the calculation of SG mass releases used to evaluate the radiological consequences of the event. The analysis begins with the plant operating at HFP with both CVCS pumps operating and letdown isolated. The tube rupture is postulated to occur in the shortest SG tube near the hot leg tube sheet to maximize break flow and to maximize the flashing fraction, thereby maximizing the radiological release to the secondary side. The staff concurs that this is the worst break location. RT is initiated by the operator 30 minutes after accident initiation. LOOP is assumed to occur coincident with RT, causing the loss of power to the CVCS pumps, RCPs, and MFW pumps. The operator is assumed to complete the initial SGTR mitigation steps within 10 minutes after tripping the reactor. These steps include closing the affected SG’s MSIV, resetting its MSRT setpoint high, isolating its EFWS and blowdown lines, starting the EFW pumps, and initiating partial cooldown of the unaffected SGs. All actions are performed in the control room.

TT occurs immediately after RT, causing the pressure in the affected SG to increase rapidly and the break flow to the affected SG to decrease. Pressure in the unaffected SG reaches the MSRT setpoint about 10 minutes after RT. The analysis assumes that the affected SG’s MSRCV fails open when the MSRT activates. This failure causes a rapid depressurization of the affected SG, and an increase in break flow, thus maximizing the radiological release. The staff concurs that this is the worst single failure.

The analysis shows that the partial cooldown of the three unaffected SGs is completed within 30 minutes after RT. At this point, the operator is assumed to start the EBS to provide sufficient

boration in the primary system and to initiate manual 50 °C/h (90 °F/h) cooldown of the unaffected SGs. The analysis shows that flow through the ruptured tube is terminated by about 6,000 seconds when operator actions have resulted in equalizing the primary and secondary pressures. The analysis was terminated at 10,000 seconds.

During a SGTR, the RCS depressurizes and DNBR is reduced. The SGTR analysis shows that the depressurization before RT is slower than that for the depressurization in the IOPSRV event discussed in Section 15.6.1 of this report. After RT, DNBR rapidly increases. The calculated minimum core exit subcooling was 6.1 °C (11 °F) and occurred approximately 30 minutes after reactor scram. The staff's conclusion for the IOPSRV event also applies to the SGTR event. The applicant calculated that SAFDLs were not challenged during the SGTR event. The staff concurs since subcooling and DNBR margin are maintained.

In RAI 34, Question 05.06.03-1, the staff requested that the applicant provide a clarification of FSAR Tier 2, Section 15.6.3.1, "Identification of Causes and Accident Description," regarding reactor tripping on high PZR pressure. In a September 24, 2008, response to RAI 34, Question 15.06.03-1, the applicant provided a description of the SGTR scenarios that led to RT on high pressurizer pressure. The staff finds the applicant's response to be acceptable since it provides the requested information, and correctly describes the sequence of events. Therefore, the staff considers RAI 34, Question 15.06.03-1 resolved.

In RAI 34, Question 05.06.03-2, the staff requested that the applicant present the results of the analyses that led to the conclusion that the case presented is the limiting case for radiological release. In a September 24, 2008, response to RAI 34, Question 15.06.03-2, the applicant described its methodology for identifying the limiting case. In essence, the methodology consists of two phases. The initial phase consists of running base hot and cold side break cases with no single failure at a number of potential initial conditions to determine a limiting initial condition for the second phase. The second phase analyzes various single failures. The initial condition selected for the single failure analyses assumes a reduced average reactor coolant temperature (T_{avg}) 324 °C (584 °F), five percent SG tube plugging, CVCS charging pumps to be available, and LOOP at RT. Based on 25 thermal hydraulic cases analyzed, the radiological consequences were then evaluated for four single failure cases to determine the limiting radiological release case. The staff finds the applicant's response to be acceptable and concurs that the limiting radiological release case was properly identified. Therefore, the staff considers RAI 34, Question 15.06.03-2 resolved.

During its review, the staff noted that the applicant had not provided the results of the SG overfill analysis. Therefore, in RAI 34, Question 15.06.03-3, the staff requested that the applicant present the results of the SG overfill analysis. In an October 31, 2008, response to RAI 34, Question 15.06.03-3, the applicant stated that the FSAR would be modified to include the results of the SG overfill analysis and provided the draft markup of the FSAR. The overfill analysis assumed the single failure of the affected SG EFW control valve failing in the open position. Consequently, EFW flow to the affected SG continues even if the SG level exceeds the EFW shutoff value. EFW flow to the affected SG is terminated when the operator closes the EFW isolation valve. The analysis assumes the operator takes this action 51 minutes after the SG tube rupture occurs. One hour after the event's initiation, the operator is assumed to begin cycling the PSRV to maintain RCS pressure equal to the affected SG pressure. This action terminates flow to the affected SG and terminates the SG overfill. The analysis shows that the maximum wide range level in the affected SG is 95 percent and occurs 14,000 seconds after the tube ruptures. The staff confirmed that Revision 1 of FSAR Tier 2, Section 15.6.3 contains

the changes mentioned above. The staff has reviewed the SG overfill analysis and determines this response to be acceptable. Therefore, the staff considers RAI 34, Question 15.06.03-3 resolved.

15.6.3.5 *Combined License Information Items*

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.6.3.6 *Conclusions*

The staff determines that the applicant has conducted the SGTR analysis using an approved methodology and has identified the limiting case for radiological release and the limiting case for SG overfill. The staff determines that the applicant's SGTR analyses are acceptable.

Section 15.0.3 of this report discusses the staff's evaluation and conclusions concerning the radiological release analysis for the SGTR event.

15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)

This event does not apply to the U.S. EPR.

15.6.5 Loss of Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

This section will be addressed at a later date.

15.7 Radioactive Release from a Subsystem or Component

15.7.1 Radioactive Gas Waste System Leak or Failure

Radioactive gas waste system leaks or failures are described in FSAR Tier 2, Section 11.3.3.6, "Radioactive Gaseous Waste System Leak or Failure."

15.7.2 Radioactive Liquid Waste System Leak or Failure

Radioactive liquid waste system leaks or failures are described in FSAR Tier 2, Section 11.2.3.6, "Radioactive Liquid Waste System Leak or Failure."

15.7.3 Postulated Radioactive Releases due to Liquid-Containing Tank Failures

Postulated radioactive releases due to liquid-containing tank failures are described in FSAR Tier 2, Section 11.2.3.7, "Postulated Radioactive Releases due to Liquid-Containing Tank Failures."

15.7.4 Radiological Consequences of Fuel Handling Accidents

The radiological consequences of fuel handling accidents are described in FSAR Tier 2, Section 15.0.3.10, “Fuel Handling Accident.”

15.7.5 Spent Fuel Cask Drop Accidents

The FSAR states that the spent fuel cask handling equipment is designed so that no cask can be dropped from more than the equivalent of 9.2 m (30 ft). Cask rupture from a dropped cask of less than 9.2 m (30 ft) has been calculated to not occur, and consequently, no radioactivity is released. Therefore, a radiological consequence calculation is not necessary. A description of the spent fuel handling equipment and procedures is provided in FSAR Tier 2, Section 9.1.4, “Fuel Handling System.”

15.8 Anticipated Transients Without Scram

15.8.1 Introduction

An anticipated transient without scram (ATWS) is an AOO followed by failure of the RT portion of the protection system.

15.8.2 Summary of Application

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.2.7, “Extra Borating System.”

FSAR Tier 2: The applicant has provided a FSAR Tier 2 system description in Section 15.8, summarized here, in part, as follows:

FSAR Tier 2, Section 15.8.1, “General Background,” states that the design features and strategies of the U.S. EPR, including the protection system, demonstrate that ATWS migration equipment is reliable and independent from sensor output to the final actuation device from the existing RT system and that design features related to diverse scram signals have been included in the U.S. EPR design. If the ATWS is the result of a failure within the protection system, an independent and diverse RT signal can be generated within the process automation system (PAS). The PAS bypasses the PS and initiates a RT. The PAS also includes logic that fulfills the ATWS requirements of 10 CFR 50.62, “Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants.” The PAS logic is designed to be independent of the PS from the sensor output to the final actuation device, and provides a diverse means to trip the reactor and the turbine, and to initiate emergency feedwater on conditions indicative of an ATWS. (The U.S. EPR is designed so that flow from the EFW system is not required for the first 30 minutes following an ATWS). The EBS is also available via manual actuation should additional negative reactivity be desired to shutdown the reactor. The FSAR also states that the mechanical blockage of the control rods when a RT signal is received is of such low probability that it is precluded from consideration as a cause of an ATWS event.

FSAR Tier 2, Section 15.8.2, “Anticipated Transients Without Scram,” qualitatively discusses the design features of the U.S. EPR that mitigate the consequences of an ATWS during those AOOs whose consequences would be made worse. These AOOs include the loss of reactor

coolant flow, loss of electrical load, TT, loss of condenser vacuum, LOOP, closure of a main steam isolation valve, and the rod cluster control assembly events.

FSAR Tier 2, Section 15.8.3, "Conclusion," states that events subject to ATWS considerations have been evaluated for potential damage to the barriers to fission product release. The FSAR concludes that the U.S. EPR design will maintain the integrity of barriers to prevent a fission product release if an ATWS event occurs. For each case assessed, the diverse trip functions and capabilities incorporated in the U.S. EPR design were determined to mitigate the adverse effects of an ATWS without compromising the integrity of the barriers to fission product release. These capabilities include having equipment from sensor output to final actuation device diverse from the RT system to automatically initiate the emergency feedwater system and initiate a TT under conditions indicative of an ATWS.

The FSAR also states that the U.S. EPR design's diverse scram system meets all regulatory requirements and that the consequences of ATWS affected AOOs are bounded by the safety analyses in Chapter 15. Therefore, the ATWS success criteria given in 10 CFR 50.46 and GDC 12, "Suppression of Reactor Power Oscillations"; GDC 14, "Reactor Coolant Pressure Boundary"; GDC 16, "Containment Design"; GDC 35; GDC 38, "Containment Heat Removal"; and GDC 50, "Containment Design Basis," are not required to be explicitly calculated.

ITAAC: The ITAAC associated with FSAR Tier 2, Section 15.8 for the EBS are given in FSAR Tier 1, Section 2.2.7, Table 2.2.7-3, "EBS ITAAC."

Technical Specifications: There are no Technical Specifications for this area of review.

15.8.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.8, "Anticipated Transients Without Scram," and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.8.

1. 10 CFR 50.62, as it relates to the acceptable reduction of risk from ATWS events via 1) inclusion of prescribed design features, and 2) demonstration of their adequacy.

2. 10 CFR 50.46, as it relates to maximum allowable peak cladding temperatures, maximum cladding oxidation, and coolable geometry.

3. GDC 12, as it relates to ensuring that oscillations are either not possible or can be reliably and readily detected and suppressed.

4. GDC 14, as it relates to ensuring an extremely low probability of failure of the coolant pressure boundary.

5. GDC 16, as it relates to ensuring that containment design conditions important to safety are not exceeded as a result of postulated accidents.

6. GDC 35, as it relates to ensuring that fuel and clad damage, should it occur, must not interfere with continued effective core cooling, and that clad metal-water reactor must be limited to negligible amounts.

7. GDC 38, as it relates to ensuring that the containment pressure and temperature are maintained at acceptably low levels following any accident that deposits reactor coolant in the containment.

8. GDC 50, as it relates to ensuring that the containment does not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment.

Acceptance criteria adequate to meet the above requirements include:

1. For evolutionary plants where the ATWS rule does not explicitly require a diverse scram system, the applicant may provide either a diverse scram system satisfying the design and quality assurance criteria specified in NUREG-0800, Section 7.2, "Reactor Trip System," or demonstrate that the consequences of an ATWS event are within acceptable values.

2. Applicants must demonstrate that the failure probability of failing the ATWS success criteria is sufficiently small, because either the criteria are met, or a diverse scram system is installed that reduces significantly the probability of a failure to scram.

3. NUREG-0460, "Anticipated Transient Without Scram for Light-Water Reactors."

15.8.4 Technical Evaluation

The staff notes that the PAS fulfills the requirements in that TT and emergency feedwater are actuated in addition to an independent RT.

The PAS is described in FSAR Tier 2, Section 7.8.1.2, "Functional Descriptions." The staff's review of the PAS and conclusions on whether this digital system meets the quality standards of safety-related equipment is contained in Chapter 7 of this report.

The staff notes that thermal-hydraulic analyses of an ATWS have not been provided nor are they required to numerically demonstrate that in the event of an ATWS a pressure increase does not exceed the ASME Service Level C limit of 22.06 MPa (3,200 psig).

The staff notes that the EBS is also available via manual actuation should additional negative reactivity be desired to shutdown the reactor in the case of an ATWS.

15.8.5 Combined License Information Items

There are no COL information items from FSAR Tier 2, Table 1.8-2 that affect this section.

15.8.6 Conclusions

The staff concludes that the U.S. EPR contains a diverse actuation system, which meets the requirements of 10 CFR 50.62.

The staff concludes that within the protection system, an independent and diverse RT signal can be generated within the PAS. The PAS provides a diverse means to trip the reactor and the turbine, and to initiate emergency feedwater on conditions indicative of an ATWS.

Therefore, the staff finds that the design meets the acceptance criteria guidance enumerated in NUREG-0800.

Therefore, the staff finds that:

1. GDC 12, as it relates to ensuring that oscillations are either not possible or can be reliably and readily detected and suppressed is met using the compliment of in-core fixed detectors and ex-core neutron detectors in the design of the plant, and by design with negative power, fuel, and moderator coefficients which suppress oscillations.

2. GDC 14, as it relates to ensuring an extremely low probability of failure of the coolant pressure boundary geometry is met by inclusion of the PAS in the U.S. EPR design which ensures that an ATWS will not result in a LOCA.

3. GDC 16, as it relates to ensuring that containment design conditions important to safety are not exceeded as a result of postulated accidents is met by inclusion of the PAS in the U.S. EPR design which ensures that an ATWS will not result in a LOCA.

4. GDC 35, as it relates to ensuring that fuel and clad damage, should it occur, must not interfere with continued effective core cooling, and that clad metal-water reaction must be limited to negligible amounts is met by inclusion of the PAS in the U.S. EPR design which ensures that an ATWS will not result in a LOCA.

5. GDC 38, as it relates to ensuring that the containment pressure and temperature are maintained at acceptably low levels following any accident that deposits reactor coolant in the containment is met by inclusion of the PAS in the U.S. EPR design which ensures that an ATWS will not result in a LOCA which may challenge containment integrity.

6. GDC 50, as it relates to ensuring that the containment does not exceed the design leakage rate when subjected to the calculated pressure and temperature conditions resulting from any accident that deposits reactor coolant in the containment is met by inclusion of the PAS in the U.S. EPR design which ensures that an ATWS will not result in a LOCA which may challenge containment integrity.

7. 10 CFR 50.62, as it relates to the acceptable reduction of risk from ATWS events via 1) inclusion of prescribed design features, and 2) demonstration of their adequacy is met by inclusion of the PAS in the U.S. EPR design.

8. 10 CFR 50.46, as it relates to maximum allowable peak cladding temperatures, maximum cladding oxidation, and coolable geometry is met by inclusion of the PAS in the U.S. EPR design which ensures that an ATWS will not result in a LOCA.

15.9 Boiling Water Reactor Stability

The BWR stability event is not applicable to the U.S. EPR.

15.10 Spent Fuel Pool Criticality and Boron Dilution Analysis

The spent fuel rack is a vendor-supplied component that is selected by the COL applicant. The selected vendor provides the criticality analysis.

Boration of the water in the spent fuel pool is not essential for maintaining the subcriticality of the stored fuel assemblies (refer to FSAR Tier 2, Section 9.1.2.2.2, “Spent Fuel Storage”). The staff’s review of spent fuel pool criticality is contained in Chapter 9 of this report.