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No figures were included in this section.

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No tables were included in this section.

15 TRANSIENT AND ACCIDENT ANALYSES

15.6.5 Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary

AREVA NP, Inc., (AREVA) has submitted a Final Safety Analysis Report (FSAR) describing the design and engineering of the U.S. EPR, as well as other information relating to the U.S. EPR design, in connection with its request to the U.S. Nuclear Regulatory Commission (NRC) for certification of the U.S. EPR design. This section documents the staff evaluation of the applicant's FSAR analyses of the U.S. EPR responses to postulated loss-of-coolant accidents (LOCAs), including long-term cooling. These analyses are used to determine the limiting safety system settings and limiting conditions for operation of the design, as well as design specifications for safety-related components and systems.

15.6.5.1 *Large-Break Loss-of-Coolant Accident*

15.6.5.1.1 Introduction

The large-break loss-of-coolant accident (LBLOCA) is a postulated accident resulting from instantaneous rupture of a reactor coolant system (RCS) pipe. A spectrum of break sizes for both double-ended guillotine break and split break types are analyzed. For the LBLOCA, the most limiting scenario is the cold-leg pipe between the reactor coolant pump (RCP) discharge and the reactor pressure vessel.

15.6.5.1.2 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 15.6.5.1, "Large Break Loss of Coolant Accident," summarized briefly here, as follows:

The FSAR states that there are three phases of an LBLOCA analysis: Blowdown; refill; and reflood. The first phase is the blowdown phase, which covers the period from the postulated RCS pipe break until the emergency core cooling system (ECCS) begins to inject borated water into the RCS. The refill phase begins at the end of the blowdown phase, and continues until fluid from the ECCS has filled the downcomer and lower plenum up to the bottom of the heated length of the fuel. The reflood phase is the last phase, which covers the period from the end of refill until the fuel is once again covered with water and the fuel cladding is cooled, terminating the accident.

In the blowdown phase, the RCS rapidly depressurizes from operating to saturation pressure in the hot RCS leg, which results in the fuel rods exceeding the critical heat flux. The critical heat flux is exceeded when the heat generated is greater than the heat that can be removed by the surrounding steam. During the refill phase, heat is primarily transferred from the hotter fuel rods to cooler fuel rods and structures by radiative heat transfer. During the reflood phase, the ECCS fluid flowing into the downcomer provides the driving head to move coolant through the reactor core. The fuel rods are then cooled and quenched by radiative and convective heat transfer as the quench front moves up the reactor core.

The analytical methodology used to analyze this event is described in ANP-10278P, “U.S. EPR Realistic Large Break Loss of Coolant Accident Topical Report,” Revision 1, January 2010. The realistic LBLOCA methodology consists of the following computer codes:

- RODEX3A, as described in ANF-90-144(P)(A), “RODEX3 Fuel Rod Thermal-Mechanical Response Evaluation Model, Volume 1, ‘Theoretical Manual,’ and Volume 2, ‘Thermal and Gas Release’,” which is used for computation of the initial fuel stored energy, fission gas release, and fuel-cladding gap conductance.

- S-RELAP5 for system thermal-hydraulic calculations. Containment backpressure calculations are performed by the ICECON module within S-RELAP5.

The FSAR states that the acceptance criteria for LBLOCA are met for the maximum peak clad temperature, total percentage of fuel cladding oxidation, amount of hydrogen generation, maintenance of coolable geometry of the reactor core, and long-term cooling, and that the radiological consequences are within the guidelines of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 100, “Reactor site criteria.” The staff’s evaluation of the radiological consequences is provided in Section 15.0.3 of this report.

ITAAC: There are no inspection, test, analysis, and acceptance criteria (ITAAC) items for this area of review.

Technical Specifications: The Technical Specifications (TS) values of Fq, Peak to average enthalpy rise (FΔh) (TS 3.2.2), and core power (TS 3.3.1) cannot be higher than the upper limit values used in the realistic large-break loss-of-coolant accident (RLBLOCA) methodology used in this FSAR section, as described in TS B 3.2.

15.6.5.1.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review and the associated acceptance criteria are given in NUREG-0800, “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 15.6.5, “Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary,” and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.6.5.

Requirements

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

1. General Design Criterion (GDC) 13, “Instrumentation and Control,” 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 35, “Emergency Core Cooling,” as it relates to demonstrating that the ECCS would provide abundant emergency core cooling (ECC) to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that

(1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts.

3. 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors,” as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a LOCA resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary (RCPB).

Acceptance criteria adequate to meet the above requirements include:

1. An evaluation of ECCS performance has been performed by the applicant in accordance with an evaluation model that satisfies the requirements of 10 CFR 50.46. Regulatory Guide (RG) 1.157, “Best-Estimate Calculations of Emergency Core Cooling System Performance,” May 1989 provide guidance on acceptable evaluation models.

   The analyses should be performed in accordance with 10 CFR 50.46, including methods referred to in 10 CFR 50.46(a)(1) or 10 CFR 50.46(a)(2). The analyses must demonstrate sufficient redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities such that the safety functions could be accomplished assuming a single failure in conjunction with the availability of onsite power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available). Additionally the LOCA methodology used and the LOCA analyses should be shown to apply to the individual plant by satisfying 10 CFR 50.46(c)(2), and the analysis results should meet the performance criteria in 10 CFR 50.46(b).

   A. The calculated maximum fuel element cladding temperature does not exceed 1200 °C (2200 °F).

   B. The calculated total local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation as well as oxidation that occurs during the course of the accident.

   C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed one percent of 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.

   D. Calculated changes in core geometry are such that the core remains amenable to cooling.

2. Three Mile Island (TMI) Action Plan Items II.E.2.3, II.K.2.8, II.K.3.5, II.K.3.25, II.K.3.30, II.K.3.31, and II.K.3.40 have been met.

15.6.5.1.4 Technical Evaluation

As described in detail below, the staff performed a technical evaluation of the applicant’s LBLOCA analysis and verified that the applicant used an approved methodology as well as approved versions of the relevant computer codes. The staff evaluated input parameters to confirm that they were reasonable or conservative, and verified that uncertainty parameters were within their appropriate ranges. The staff reviewed the applicant’s results for consistency and reasonableness and to verify that they met the regulatory acceptance criteria. The staff issued requests for additional information (RAIs) and performed selective independent calculations to clarify certain aspects of the applicant’s analysis.

The applicant used the methodology described in ANP-10278P, “U.S. EPR Realistic Large Break Loss of Coolant Accident Methodology,” to conduct realistic LBLOCA analysis. The applicant submitted topical report ANP-10278P, “U.S. EPR Realistic Large Break Loss of Coolant Accident Methodology,” Revision 1, which describes the application of the applicant’s RLBLOCA methodology, with modifications, to the U.S. EPR. Topical report ANP-10278P has been submitted but has not yet been approved by the NRC. The review of the U.S. EPR realistic large-break loss-of-coolant accident methodology is contained in the staff’s safety evaluation report on ANP-10278P. The staff issued RAI 493, Question 15.06.05-98 to track the ongoing review of ANP-10278P. **RAI 493, Question 15.06.05-98, is being tracked as an open item.**

The applicant’s RLBLOCA methodology is based upon nonparametric statistics. A spectrum of break size and break type (double-ended guillotine break (DEGB) or split break) is analyzed for both the first core cycle and an equilibrium reload cycle. Each break spectrum is analyzed by making 124 S-RELAP5 simulations. In each of these simulations, 26 out of 28 key LOCA parameters (18 phenomenological and 10 plant-related) is randomly sampled over a range (core power level and the availability of offsite power are not sampled, as discussed below). Each parameter’s range has been established by code uncertainty assessments or by expected operational limits. The staff notes that the analysis is not conducted by varying break size alone. Since 26 parameters are perturbed for each S-RELAP5 simulation, it is not possible to compare any two of the 124 cases for the purpose of quantifying the sensitivity of peak cladding temperature (PCT) to variations of a single input parameter. The application of the RLBLOCA methodology to the U.S. EPR does not allow sampling of core power; the initial power is set to the nominal full power value plus measurement uncertainty. Offsite power availability is also not sampled in the U.S. EPR application; rather, loss of offsite power (LOOP) is assumed. This is conservative, because it leads to the longest delay to safety injection (SI). The U.S. EPR is designed so that the RCPs trip almost immediately on low delta-P (low differential pressure from pump inlet to outlet) signals in the event of a LOCA. Therefore, LOOP and non-LOOP cases are essentially the same with respect to RCP trip time.

Three safety parameters are obtained from the RLBLOCA analysis: The PCT; the maximum total local cladding oxidation; and the maximum core wide cladding oxidation. The PCT, local oxidation, and core wide oxidation are extracted from each of the 124 S-RELAP5 cases and gathered into three lists of safety parameter values. The maximum value is then taken from each list for comparison to the corresponding regulatory criterion. The theory of non-parametric statistics is that if these three maximum values conform to the acceptance criteria, then the plant meets 10 CFR 50.46 at the 95/95 tolerance level, and is therefore acceptable.

Although the applicant’s RLBLOCA methodology is realistic, it still incorporates the following conservatisms:

- The worst single failure, which is failure of an emergency diesel generator (EDG) for the ECCS of an unbroken loop, is incorporated into the analysis as an initial condition.

In addition, another ECCS train, randomly chosen from the two remaining unbroken loops, is assumed out of service for preventative maintenance. This assumption is conservative because it reduces the amount of SI available during the postulated event.

- Minimum delivery curves are used for medium-head safety injection (MHSI) and low-head safety injection (LHSI) pumps. This assumption is conservative because it reduces the amount of SI injected during the postulated event.

- No credit is taken for reactor SCRAM. This assumption is conservative because it does not take credit for the negative reactivity associated with the control rods.

- Safety injection system (SIS) actuation setpoints are set to conservative values including consideration of measurement uncertainty. This assumption is conservative because it maximizes the time delay to SI.

- Containment volume is sampled over a range bounded by the maximum estimate of free volume of the containment including internals to the volume of an empty containment. Maximizing the containment volume results in a lower calculated back pressure to the RCS during reflood. This is conservative because it maximizes the steam binding in the RCS loops, which inhibits reflood, and it minimizes the steam density, which reduces the cooling capacity of the steam during reflood.

The staff has verified that the conservative assumptions listed above are part of the methodology. In ANP-10278P, the applicant determined that the limiting PCT LBLOCA case was Case 38 for the equilibrium cycle analysis and Case 85 for the initial cycle analysis, with PCTs of 885 °C (1,625 °F) and 924 °C (1,695 °F), respectively. As reported in ANP-10278P, the maximum local oxidation was given by Case 38 for the equilibrium cycle and by Case 70 for the initial cycle. The maximum total oxidation was given by Case 2 for the equilibrium cycle and Case 63 for the initial cycle as discussed in ANP-10278P. The staff's review of ANP-10278P is provided in the safety evaluation report for ANP-10278P.

To review the sampling process, the staff requested the actual values for each sampled parameter in a representative set of LBLOCA simulations. In RAI 30, Question 15.06.05-14, the staff requested that the applicant provide the values of the sampled parameters and the range of each parameter used for the 59 cases for the equilibrium cycle and the 59 cases for the initial cycle analyses, including (RG the PCT result for each case, and providing the information in an Excel spreadsheet format, if possible.

In an August 28, 2008, response to RAI 30, Question 15.06.05-14, the applicant provided two spreadsheets containing all the sampled parameters' values. The staff reviewed the sampled parameter values transmitted in the spreadsheets and verified that all of the relevant parameters were included and they were within their specified ranges. The staff finds that the applicant's response is therefore acceptable.

In RAI 30, Question 15.06.05-15, the staff requested that the applicant clarify the information provided in several figures in FSAR Tier 2, Section 15.6, "Decrease in Reactor Coolant Inventory Events," by confirming that FSAR Tier 2, Figures 15.6-27 through 15.6-36 were from the same simulation as FSAR Tier 2, Figures 15.6-37 through 15.6-50. In an August 28, 2008, response to RAI 30, Question 15.06.05-15, the applicant confirmed that the figures were from the same simulation.

During the staff's review of FSAR Tier 2, Section 15.6.5, the staff questioned whether the overspeed of the broken loop pump was realistic and asked what the effect of having a locked rotor would be instead. Therefore, in RAI 30, Question 15.06.05-17, the staff requested that the applicant provide the methodology used to determine that the overspeed stated in FSAR Tier 2, Figure 15.6-49 can be physically accommodated. In an August 28, 2008, response to RAI 30, Question 15.06.05-17, the applicant stated that the evaluation of RCP overspeed for the U.S. EPR does not need to assume a DEGB in the main coolant piping, since leak before break (LBB) is the basis for overspeed evaluation. The staff's evaluation of using LBB as the basis for pump overspeed evaluation is given in Section 3.6.3 of this report. The applicant's response also noted the RLBLOCA methodology assumes that the pump will free-wheel in the S-RELAP5 simulation of the RLBLOCA. The applicant stated that an evaluation assuming that the broken loop RCP rotor will lock instead of free-wheel was performed and showed no more than a 19 °C (35 °F) increase in PCT. The review of the U.S. EPR realistic large-break loss-of-coolant accident methodology is contained in the staff's safety evaluation report on ANP-10278P. The staff issued RAI 493, Question 15.06.05-98, to track the ongoing review of ANP-10278P. **RAI 493, Question 15.06.05-98, is being tracked as an open item.**

In RAI 167, Question 15.06.05-38, the staff requested that the applicant provide RODEX3A predicted hot rod centerline and fuel average temperatures as a function of burnup, so the staff could compare them to its own calculations. In a March 31, 2009, response to RAI 167, Question 15.06.05-38, the applicant supplied the requested information. Therefore, the staff finds the applicant's March 31, 2009, response to RAI 167, Question 15.06.05-38, acceptable. Subsequent comparisons to the staff's calculations resulted in a revision of the applicant's treatment of initial store energy in its LBLOCA methodology. The details of that revision and the staff's review of it are documented in the Section 4.2.14 of the SER for ANP-10278. The staff's confirmatory calculations are discussed below.

In RAI 167, Question 15.06.05-41, the staff requested that the applicant provide an explanation of how the initial core stored energy is accounted for in the analyses of the U.S. EPR. In a March 31, 2009, response to RAI 167, Question 15.06.05-41, the applicant described how the initial fuel conditions are determined for non-LOCA event analyses and why they are conservative. The applicant noted that portions of its response were incorporated into FSAR Tier 2, Section 15.0.0.3.1, "Design Plant Conditions and Initial Conditions." The stored energy is not important in SBLOCA and non-LOCA events for the U.S. EPR design because there is sufficient time for heat transfer to the coolant during the transient. Accordingly, the staff finds the applicant's response acceptable and considers RAI 167, Question 15.06.05-41, resolved.

Confirmatory Calculations

The staff conducted confirmatory LBLOCA calculations using both RELAP5 and TRACE. These simulations used a top skewed power profile with the peak power location having an F_q of 2.6 (Technical Specification limit), no steam generator (SG) tube plugging, and design RCS flow. These assumptions are conservative because they maximize inventory loss and maximize PCT. The staff assumed one SI train out of service due to maintenance and another unavailable due to failure of an EDG.

Both RELAP5 and TRACE simulations conducted by the staff identified the DEGB with unity break flow multipliers (i.e., 100 percent of the theoretical break flow, which maximizes inventory loss and maximizes PCT) as the limiting accident scenario. Calculated PCTs ranged from 677 °C to 787 °C (1,250 °F to 1,448 °F). The two limiting cases presented by the applicant had

PCTs of 774 °C (1,425 °F) and 883 °C (1,531 °F). The staff's calculations indicate that for the LBLOCA there is a substantial core performance safety margin in the U.S. EPR. The staff's calculation showed that the applicant's LBLOCA modeling is conservative.

The staff compared the staff's confirmatory calculations to the applicant's limiting case (Case 38 provided the most limiting PCT). Based on this comparison, the staff raised questions in the review of ANP-10278P regarding core flow oscillations and calculated pressure response when the accumulator cover gas is released into the loops. The staff evaluation of these issues is presented in Sections 4.2.4.2 and 4.2.4.3 of the SER on ANP-10278P. The staff notes that the final safety evaluation report (FSER) on ANP-10278P is not yet complete. The staff issued RAI 493, Question 15.06.05-98, to track the ongoing review of ANP-10278P. **RAI 493, Question 15.06.05-98, is being tracked as an open item.**

15.6.5.1.5 Combined License Information Items

There are no COL information items related to this area of review. The staff determined that no COL information items need to be included in FSAR Tier 2, Table 1.8-2, "U.S. EPR Combined License Information Items," for LBLOCA consideration.

15.6.5.1.6 Conclusions

Except for the open item identified above relating to the ongoing review of ANP-10278P, the staff concludes that:

- 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 is met by assuring that SIS actuation setpoints are set to conservative values including consideration of measurement uncertainty.

- GDC 17, as it relates to onsite and offsite electric power systems, to ensure that SSCs function during normal operation, including anticipated operational occurrences (AOOs) is met in that the analysis was conservatively performed assuming loss of offsite power.

- GDC 26, as it relates to the control of reactivity changes so SAFDLs are not exceeded during AOOs is met in that credit for control rod insertion was not taken in the LBLOCA analysis.

- GDC 35, as it relates to demonstrating that the ECCS would provide abundant ECC to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts.

- 10 CFR 50.46, as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a LOCA resulting from a spectrum of postulated piping breaks within the RCPB is met in that the analysis shows that the acceptance criteria are not exceeded.

- 10 CFR 52.47(a) , as it relates to demonstrating compliance with any technically relevant portions of the TMI-related requirements set forth in 10 CFR 50.34(f)(1)(vi) and

10 CFR 50.34(f)(1)(iii), for the U.S. EPR in that the analysis was performed with consideration of these items.

- 10 CFR Part 100, as it relates to mitigating the radiological consequences of an accident is met as described in Section 15.0.3 of this report.

15.6.5.2 *Small-Break Loss-of-Coolant Accident*

15.6.5.2.1 Introduction

The small-break loss-of-coolant accident (SBLOCA) is a postulated accident resulting in a break in a pipe carrying reactor coolant that has an area of less than approximately 10 percent of the RCS cold leg pipe cross-sectional area. This range of pipe break area encompasses the small lines that penetrate the RCPB and generally represents small piping lines such as relief and safety valve lines, charging and letdown lines, drain lines, and instrumentation lines.

15.6.5.2.2 Summary of Application

FSAR Tier 1: The FSAR Tier 1 information associated with this section is found in FSAR Tier 1, Section 2.8.2, "Main Steam System," for the main steam relief trains (MSRTs).

FSAR Tier 2: The applicant has provided an FSAR Tier 2, Section 15.6.5.2, "Small Break Loss of Coolant Accident," system description summarized here, in part, as follows:

The FSAR states that the SBLOCA results in a loss of reactor coolant inventory that cannot be offset by injection from the chemical volume and control system (CVCS), which is assumed to be unavailable, since it is not a safety-related system. The loss of primary reactor coolant causes a decrease in the RCS pressure and pressurizer level and leads eventually to a reactor trip, turbine trip, and SI. With activation of the ECCS, including SI, the RCS is cooled down and depressurized. The SBLOCA results in the secondary side SG pressure increasing and, because of the assumed unavailability of the steam dump to the main condenser, the MSRTs open to relieve steam to the atmosphere and cool down the SGs.

The subsequent change in the RCS water inventory depends on the balance between the ECCS flow rates and the pipe break flow rate. A portion of the reactor core may be uncovered before the rate of ECCS water addition exceeds the loss of RCS coolant out the break. If this occurs, the fuel clad temperature will rise above saturation temperature in the uncovered part of the core.

The SBLOCA cases are analyzed until the top of the active fuel is recovered with a two-phase mixture and the cladding temperatures are reduced to temperatures near the saturation temperature. The SBLOCA analyses are performed using the methodology documented in the approved topical report ANP-10263P-A, "Codes and Methods Applicability Report for the U.S. EPR," August, 2007. Technical Report ANP-10291P, "Small Break LOCA and Non-LOCA Sensitivity StudiesSudies and Methodology," October 2007, describes SG nodalization sensitivity analyses performed to support the SBLOCA methodology. The FSAR states that the appropriate conservatisms, prescribed by 10 CFR Part 50, Appendix K are incorporated in these analyses.

The computer codes used in this analysis are:

- The RODEX2-2A computer code, as described in RODEX2, “Fuel Rod Thermal-Mechanical Evaluation Model,” August 1986 is used to calculate the burnup dependent initial fuel conditions for each active core region in S-RELAP5.
- The S-RELAP5 computer code as described in EMF-2328(P)(A), “PWR Small-Break LOCA Evaluation Model, S-RELAP5 Based,” March 2001, is used to model the primary system (including the hot fuel rod) and the secondary side of the SGs.

The applicant’s FSAR analysis concludes that: (1) The limiting SBLOCA case is the 0.165 m (6.5 in.) cold leg break at the reactor coolant discharge piping with a loss of offsite power at the time of reactor trip; (2) the acceptance criteria for the SBLOCA are met for the maximum peak clad temperature, total percentage of fuel cladding oxidation, the amount of hydrogen generation, maintaining coolable geometry of the reactor core, and long-term cooling; and (3) the radiological consequences are described in Section 15.0.3 of this report.

ITAAC: The ITAAC associated with FSAR Tier 2, Section 15.6.5.2 are given in FSAR Tier 1, Section 2.8.2 for the MSRTs.

Technical Specifications: The Technical Specifications values of Fq, FΔh, and core power cannot be higher than the upper limit values used in the RLBLOCA methodology used in this FSAR section.

15.6.5.2.3 Regulatory Basis

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.6.5 and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.6.5.

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 35, as it relates to demonstrating that the ECCS would provide ECC to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts.
3. 10 CFR 50.46, “Acceptance criteria for emergency core cooling systems for light water nuclear power reactors,” as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a LOCA resulting from a spectrum of postulated piping breaks within the RCPB.

Acceptance criteria adequate to meet the above requirements include:

1. An evaluation of ECCS performance has been performed by the applicant in accordance with an evaluation model that satisfies the requirements of 10 CFR Part 50, Appendix K, in that the SBLOCA analysis uses the assumptions in this appendix.

The analyses must demonstrate sufficient redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities such that the safety functions could be accomplished assuming a single failure in conjunction with the availability of onsite power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available). Additionally the LOCA methodology used and the LOCA analyses should be shown to apply to the individual plant by satisfying 10 CFR 50.46(c)(2), and the analysis results should meet the performance criteria in 10 CFR 50.46(b).

A. The calculated maximum fuel element cladding temperature does not exceed 1,204 °C (2,200 °F).

B. The calculated total local oxidation of the cladding does not exceed 17 percent of the total cladding thickness before oxidation. Total local oxidation includes pre-accident oxidation as well as oxidation that occurs during the course of the accident.

C. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed one percent of 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.

D. Calculated changes in core geometry are such that the core remains amenable for cooling.

2. The most limiting plant systems single failure as defined in the, “Definitions and Explanations,” of 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plants,” 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,” November 2003.

3. The TMI Action Plan requirements for II.E.2.3, II.K.2.8, II.K.3.5, II.K.3.25, II.K.3.30, II.K.3.31, and II.K.3.40 have been met.

15.6.5.2.4 Technical Evaluation

The staff’s technical evaluation of the applicant’s SBLOCA analysis verified that the applicant used an approved methodology and approved versions of the computer codes the applicant employed in its analysis. The staff evaluated the applicant’s input parameters to confirm that they were reasonable or conservative and verified adherence to the requirements of 10 CFR Part 50, Appendix K. The staff reviewed the applicant’s results for consistency and reasonableness and to verify that they met the regulatory acceptance criteria. The staff issued RAIs and performed selective independent calculations to clarify several aspects of the applicant’s analysis.

The applicant has analyzed a spectrum of SBLOCAs using its approved methodology for operating pressurized-water reactors (PWRs) as described in EMF-2328(P)(A). The application of this methodology to the U.S. EPR was approved by the NRC (Final SER for Topical Report ANP-10263, “Codes and Methods Applicability Report for the U.S. EPR,” August 2007) subject to the applicant performing SG nodalization sensitivity calculations and NRC approval of same. In November 2007, the applicant submitted the requested SG nodalization calculations in ANP-10291. The staff's technical evaluation of that report, as it pertains to the SBLOCA, is provided below.

The applicant's analyses incorporate a number of conservative assumptions. The reactor power is assumed to be 100.5 percent of nominal to account for measurement uncertainty. The staff's evaluation of the 0.5 percent power measurement uncertainty is contained in Section 15.0 of this report. Decay heat at 120 percent of the American Nuclear Society (ANS) 5.1-1973 Standard, "Decay Heat Estimates for MNR," is assumed, as required by 10 CFR Part 50, Appendix K. For the purpose of analysis, the hot rod and the hot assembly power are assumed to be at the FΔh Technical Specification limit. This assumption is conservative because it prevents the hot rod from radiating heat to surrounding rods, which are normally cooler. Two safety trains are assumed to be unavailable, one due to a single failure of an EDG, and another due to preventive maintenance. One of two available trains is assumed to belong to the broken loop, thus increasing the loss of ECC to the break. Two loop seals are assumed to be biased in the analysis as deeper than would normally be the case during actual operation. The loop seal in the broken loop and the one in an adjacent loop are each deepened by 0.3048 m (1 ft) so that clearing of these loop seals is delayed, which delays the establishment of natural circulation cooling. Sensitivity studies have shown that higher PCTs result when clearing of the broken loop's loop seal is delayed. Finally, the analysis assumes a bounding top skewed axial power profile with the peak power location at a normalized distance of 0.8542. The peaking factor at the peak power location is set to the Technical Specification F_q limit.

Two break spectra (5 cm (1.969 in.) to 20 cm (7.87 in.) in diameter in steps of 1.3 cm (0.511 in.)) were analyzed and reported: One assumes a LOOP coincident with reactor SCRAM; the other assumes offsite outside power is available. In the latter case, the RCPs trip automatically when a low differential pressure (DP) is detected across two of the four RCPs in conjunction with an SI signal.

The break spectrum for the SBLOCA with LOOP predicts that PCTs remain below 660 °C (1,220 °F) for break sizes smaller than 15 cm (6 in.) in diameter. The peak predicted PCT of 892 °C (1,638 °F) occurred for the 16.5 cm (6.5 in.) break. PCTs for larger breaks were predicted to be 28 °C to 89 °C (50 °F to 160 °F) lower.

The break spectrum for the SBLOCA without LOOP predicts that PCTs remain between 760 C (1,400 °F) and 871 °C (1,600 °F) for break sizes ranging from 8.9 to 20.3 cm (3.5 in. to 8 in.) The higher PCTs predicted for the smaller breaks for the non-LOOP SBLOCA are due to the methodology's prediction that more liquid will spill out the break by running RCPs longer. As break size increases, the RCPs trip sooner, and the PCTs for the non-LOOP cases approach those for the LOOP cases.

The 16.5 cm (6.5 in.) SBLOCA with LOOP was identified as the limiting SBLOCA case, with a predicted PCT of 892 °C (1,638 °F). This case is characterized by a heatup predicted to begin at 150 seconds (sec) from the time of the break, and ending at 360 sec. The methodology predicts that heatup begins because a large amount of liquid is modeled to eject from the system between 100 sec and 200 sec following the break. The methodology predicts that the loop seals in Loops 2 and 3 clear at 235 sec, and further predicts that steam will reach the break and cause a large increase in the primary system's depressurization rate. The methodology predicts that pressure falls to the accumulators' actuation pressure by 360 sec. The methodology then predicts that a large inflow of liquid from the accumulators terminates the core heatup at 360 sec. The methodology predicts that LHSI begins to inject at 380 sec and is sufficient to keep the core covered thereafter. Only a small amount (0.38 percent) of cladding oxidation is calculated to occur.

The applicant's break flow calculation shows break flow increases by 136 kg/s (300 lbm/s) during the 100 to 200 sec time period for no apparent reason. The cause for the predicted break flow increase was not addressed by the applicant in the FSAR. While it is apparent that the flow increase is conservative, without knowing why the S-RELAP5 model predicts the flow increase the staff remained concerned that there might be a problem with the model. Therefore, in RAI 147, Question 15.06.05-26b, the staff requested that the applicant explain the reason for the break flow increase. In a January 9, 2009, response to RAI 147, Question 15.06.05-26b, the applicant explained that the jump in break flow was due to break model limitations in S-RELAP5. The break flow model for subcooled flow is a realistic model, while the model for saturated break flow is not; it is the conservative Moody model required by 10 CFR Part 50, Appendix K for the conservative methodology described there. As a consequence, there is a physically unrealistic increase in break flow predicted as the fluid at the break is modeled to begin to flash. The staff finds the applicant's explanation acceptable, because it satisfactorily explains the predicted break flow increase and the flow increase is in a conservative direction. Therefore, the staff considers RAI 147, Question 15.06.05-26b, resolved.

The predicted limiting size for the SBLOCA changed from 10.2 cm (4 in.) in ANP-10263 to 16.5 cm (6.5 in.) in the FSAR. In RAI 30, Question 15.06.05-20, the staff requested that the applicant explain what model assumptions resulted in this change of the limiting break size. In an October 21, 2008, response to RAI 30, Question 15.06.05-20, the applicant stated that to respond to the request, the applicant ran a series of SBLOCA simulations and systematically removed the changes made to the S-RELAP5 model between the time of the ANP-10263 analysis and the FSAR analysis. A break spectrum was run each time a model change was removed. The break spectrum ranged from 5.1 cm (2 in.) to 20.3 cm (8 in.). This is a much larger range than was used in the ANP-10263 analysis. With the broader break spectrum, the analysis showed that when the S-RELAP5 model was restored to that used in the ANP-10263, the 16.5 cm (6.5 in.), not the 10.2 cm (4 in.) break, was the limiting break. The apparent inconsistency between the ANP-10263 and the FSAR was due to the applicant's narrow break spectrum range used in ANP-10263. The applicant's October 21, 2008, response to RAI 30, Question 15.06.05-20, demonstrated that the limiting break size for every model change was either 16.5 cm (6.5 in.) or 17.8 cm (7 in.) and the PCT for every model change was below the value reported in the FSAR. Accordingly, the staff finds the applicant's explanation acceptable and considers RAI 30, Question 15.06.05-20, resolved.

The applicant conducted SG nodalization studies as presented in technical report ANP-10291P to verify the adequacy of SG nodalization for SBLOCA analyses. SBLOCA calculations were conducted using three models: (1) The base model; (2) a model with double the number of axial nodes along the SG tubes; and (3) a model with the same axial noding as the base model but with three parallel paths, instead of one, representing the SG tubes.

ANP-10291P contains information associated with the timing of termination of the temperature rise associated with the limiting SBLOCA. Based on the review of the information in ANP-10291P, the staff finds that the calculated PCT is sensitive to the timing of accumulator actuation. This fact must be considered along with the overall conservatism of the SBLOCA methodology when assessing the results of the SG nodalization sensitivity studies performed by the applicant and reported in ANP-10291P and in the September 24, 2008, response to RAI 30, Question 15.06.05-21, below.

The applicant described the results of the SG nodalization studies in relation to changes in radial nodalization in ANP-10291P. In ANP-10291P, the applicant concluded that the radial noding of the SG tubes in the base model is adequate. Based on the review of the proprietary information contained in ANP-10291P related to radial nodalization sensitivity studies, the staff concurs.

The staff determined that the results of the SG axial nodalization sensitivity study were not adequately addressed by the applicant in ANP-10291P. Based on the results provided in Table 15.06.05-21-1, "Non-LOOP Break Spectrum Results", in the September 24, 2008, response to the RAI 30, Question 15.06.05-21, the calculated PCT change due to the SG axial re-nodalization was maximized for the 14 cm (5.5 in.) break and was 115 °C (207 °F). ANP-10291P contains sensitivity studies on changing break size and the cases that produce limiting results for the LOOP and non-LOOP cases, respectively. The applicant asserted in ANP-10291P that the base model is adequate. In the staff's opinion, adequate technical rationale for this assertion was not provided in ANP-10291P. Therefore, in RAI 30, Question 15.06.05-21, and RAI 147, Question 15.06.05-26a, the staff requested that the applicant address its concerns.

In a September 24, 2008, response to RAI 30, Question 15.06.05-21 and a January 29, 2009, response to RAI 147, Question 15.06.05-26a, the applicant explained that they conducted a "mini-SBLOCA spectrum analysis" using the detailed SG axial nodalization model presented in ANP-10291P. This model contains more nodes along the length of the steam generator tube to better model heat transfer across the steam generator. The results showed that the limiting case for the non-LOOP break spectrum using the detailed SG nodalization changed from 15.2 cm (6.0 in.) to 16.5 cm (6.5 in.); however, it still remained in the 14 cm (5.5 in.) to 16.5 cm (6.5 in.) region and it did not switch to a smaller size break. The applicant noted that increasing the number of nodes in the SG did increase the limiting case's PCT, but this increase was due to a predicted delay in loop seal clearing time rather than a direct result of the SG nodalization changes. Therefore, the applicant concluded that the S-RELAP5 model's original (few axial SG nodes) was adequate. The staff agreed that the increased PCT was due to a delay in loop seal clearing predicted by the model and not due to nodalization of the steam generator tubes. The staff agrees with the applicant's explanation and concurs that the existing axial nodalization of the SG in S-RELAP5 is sufficient. Accordingly, the staff considers RAI 30, Question 15.06.05-21, and RAI 147, Question 15.06.05-26a, resolved.

In RAI 147, Question 15.06.05-26c, the staff requested that the applicant explain why its calculated PCT did not change smoothly as break size increased. Instead of monotonically increasing to a maximum and then decreasing as break size is increased, the calculated PCT is characterized by peaks and valleys. For example, the PCT is 655 °C, 593 °C, and 649 °C (1,210 °F, 1,100 °F, 1,200 °F) for the 12.7 cm, 14.0 cm, and 15.2 cm (5.0 in., 5.5 in., and 6.5 in.) breaks, respectively. In an April 9, 2009, response to RAI 147, Question 15.06.05-26c, the applicant agreed that the PCT of the LOOP cases varies approximately plus or minus 55 °C (100 °F), which is about an averaged monotonic trend versus break size. It was noted that this behavior was not unique to S-RELAP5 calculations. Two other SBLOCA break spectrum calculation results were shown, one by Argonne National Laboratory, and one by EG&G, Idaho.

Both of the calculations were done using RELAP5 and both showed a non-monotonic trend in PCT with increasing break size.

The applicant explained that the PCT calculated in an SBLOCA analysis is controlled by the extent of core inventory depletion modeled and the time and amount of modeled SI. Two phenomena, reflux condensation (condensation of steam on the SG tubes hot side and the return of the condensate to the core via the hot legs) and loop seal clearing, affect the predicted core inventory in a non-uniform manner as break size changes. Furthermore, the modeled injection of cold emergency feedwater (EFW) in two of the four SGs creates an asymmetrical condition that contributes to the non-uniform influence of reflux condensation and loop seal clearing. The applicant noted that the PCT versus break size trend for cases without LOOP, in which the model does not initiate the EFW system are much smoother than the LOOP cases. The applicant concluded that PCT variability was caused by phenomena that are discontinuous with break size. The staff finds the applicant's April 9, 2009, response to RAI 147, Question 15.06.05-26c, acceptable, but believes that the PCT variability is not caused by phenomena that are discontinuous with break size but rather by the discontinuous behavior of the models of the phenomena with changes in flow regimes and SG tube heat transfer. For example, a switch from annular-mist to mist flow can cause a large change in the calculated condensation rate on the SG tubes. The discretization inherent to the calculational model also plays a role. Nonetheless, the PCT variability of 55 °C (99 °F) is much smaller than the conservatism in the calculated PCT due to 10 CFR Part 50, Appendix K assumptions. Accordingly, the staff finds the variability in PCT acceptable and, therefore, considers RAI 147, Question 15.06.05c, resolved.

In RAI 167, Question 15.06.05-28, the staff requested that the applicant provide an explanation of the RCP trip logic and operator actions credited during an SBLOCA. In a March 31, 2009, response to RAI 167, Question 15.06.05-28, the applicant explained that the U.S. EPR has an automatic RCP trip, and no operator action is needed with respect to tripping the RCPs. The trip utilizes the differential pressure (ΔP) across the RCPs to indicate the presence of two-phase flow. When one of two ΔP measurements from two of the four RCPs is below the setpoint and an SI signal is present, the RCPs will trip. The trip setpoint is 80 percent of the initial ΔP across the pump. The dependence of the RCP trip function on the presence of an SI signal reduces the possibility of a spurious RCP trip. The dependence of the RCP trip on the SI signal also prevents the RCPs from automatically tripping during a steam generator tube rupture (SGTR) event without LOOP, which is advantageous for SGTR as discussed in Section 15.6.3 of this report.

In RAI 167, Question 15.06.05-28, the staff requested that the applicant explain how RCP seals are protected during an SBLOCA. In a March 31, 2009, response to RAI 167, Question 15.06.05-28, the applicant explained that during normal operation, the RCP seal injection water is supplied by the CVCS. During an SBLOCA without LOOP in which a stage one containment isolation signal is generated, the CVCS letdown line is isolated but the RCP seal injection flow is maintained. If a stage two containment isolation signal is generated, the seal injection flow is isolated. In this case, the component cooling water system provides cooling to the thermal barrier of the RCP seals. Accordingly, pump seal integrity is maintained and a SBLOCA will not mechanistically lead to a larger break.

In RAI 167, Question 15.06.05-39, and RAI 167, 15.06.05-42, the staff requested that the applicant clarify how core stored energy is accounted for in the SBLOCA analysis. RAI 167, Question 15.06.05-39, also requested that the applicant provide an explanation of how burnup effects are accounted for in the SBLOCA analysis. In a March 31, 2009, response to RAI 167, Question 15.06.05-39, the applicant explained that fuel rod initial conditions are calculated at end-of-cycle (EOC) exposure using RODEX2-2A. Since RODEX2-2A does not account for the degradation of fuel thermal conductivity with burnup, it under-predicts the fuel initial stored energy at EOC conditions. However, the effect of the fuel stored energy on SBLOCA results is negligible, according to the applicant, because it is dissipated prior to the occurrence of PCT as the response to RAI 167, Question 15.06.05-42, demonstrates. The staff concurs, because the characteristic times of the fuel rod heat transfer are less than 10 sec, while the SBLOCA is of the order of hundreds of sec.

RAI 167, Question 15.06.05-42, requested that the applicant increase the fuel centerline temperatures for the hot rod by 222 °C (400 °F), the hot assembly by 194 °C (350 °F), the inner region fuel by 111 °C (200 °F), and the outer region fuel by 55 °C (100 °F) and examine the effect of this stored energy increase upon PCT for the larger SBLOCA break sizes. In a March 31, 2009, response to RAI 167, Question 15.06.05-42, the applicant responded with two SBLOCA calculations (20.3 cm (7.99 in.) and 24.7 cm (9.72 in.) break diameter) done at beginning of cycle (BOC) rather than EOC fuel conditions. The applicant noted that the use of BOC fuel conditions resulted in fuel centerline temperatures that were 140 °C – 222 °C (250 °F - 400 °F) higher than the values requested by the staff. Comparison of the new SBLOCA simulations with the original ones having less stored energy showed no significant change in either PCT or thermal-hydraulic response. The PCT for the 20.3 cm (7.99 in.) breaks occurred with 7 sec of one another and the higher stored energy case had a PCT, which was 11 °C (20 °F) above the low stored energy case. For the 24.7 cm (9.72 in.) break cases, PCTs occurred within 5 sec of one another with the higher stored energy case having a PCT that was 10 °C (18 °F) above the low stored energy case. The applicant concluded that its results demonstrate that the SBLOCA methodology's treatment of fuel rod stored energy is sufficient. The staff agrees that large changes in initial fuel pellet temperatures have a very small effect on calculated PCTs as described above. Therefore, the staff finds the responses to RAI 167, Questions 15.06.05-39 and 15.06.05-42, acceptable.

In the FSAR, the applicant assumed the failure of an EDG as the single worst equipment failure. In RAI 30, Question 15.06.05-22, the staff requested that the applicant provide justification for this assumption. In particular, the staff requested that the applicant show how the failure of an EDG compared to the failure of the main steam relief control valve (MSRCV) in the broken loop SG. In a September 24, 2008, response to RAI 30, Question 15.06.05-22, the applicant provided the results of simulations with each of the two postulated single failures. Each of the simulations was for the limiting SBLOCA case, a 16.5 cm (6.5 in.) break in a RCP discharge line, with LOOP assumed at reactor trip. The sensitivity analysis assumed that the MSRT in SG-4 (broken loop) is unavailable for heat removal due to the failure of the main steam relief isolation valve (MSRIV) to open. The failure of the MSRIV to open is assumed instead of the MSRCV failing closed. This assumption is bounding, because the MSRCV is normally opened, and it will allow some energy removal before it closes, whereas the MSRIV is normally closed. Therefore, for this sensitivity case, three SI trains and three EFW trains are available for recovery (with one EDG in preventive maintenance). A PCT of 843 °C (1,551 °F) was calculated for the MSRT-4 SF case, which is bounded by the PCT of 892 °C (1,638 °F), obtained for the one EDG SF case. In view of these calculations, the staff finds the September 24, 2008, response to RAI 30, Question 15.06.05-22, acceptable and agrees that failure of the EDG is the limiting single failure. Accordingly, the staff considers RAI 30, Question 15.06.05-22, resolved.

The S-RELAP5 SBLOCA simulations isolate the accumulators when non-condensable (NC) gas is detected at each accumulator's exit nozzle. The applicant's stated reason for this input assumption is that S-RELAP5 uses the HEM model to compute break flow if NC is present at the break, but 10 CFR Part 50, Appendix K requires that the Moody critical flow model be used to calculate break flow. Therefore, in RAI 30, Question 15.06.05-23, the staff requested that the applicant justify the conservatism of the accumulator isolation assumption, particularly with respect to long term cooling. In a September 24, 2008, response to RAI 30, Question 15.06.05-23, the applicant provided the results of S-RELAP5 simulations in which NC gas was allowed to enter the RCS loops. The simulations showed that the NC gases, which are modeled to enter the primary system when the accumulators empty, would travel from the SIS injection locations around the downcomer annulus and out the break without being entrained in the SG tubes. Thus, the short and long-term cooling for the SBLOCA cases would not be impacted by the potential penetration of NC into the primary system. Based on the simulations showing that non-condensable gas will not be entrained in the SG tubes, the staff finds the applicant's accumulator isolation as a modeling technique acceptable and considers RAI 30, Question 15.06.05-23, resolved.

The SBLOCA calculations presented by the applicant assume no leakage paths between the hot legs and the vessel downcomer. In RAI 30, Question 15.06.05-24, the staff requested that the applicant justify this assumption and discuss its impact on the analysis and related safety conclusions. In an October 21, 2008, response to RAI 30, Question 15.06.05-24, the applicant explained they performed a break spectrum analysis which modeled the leakage between the U.S. EPR downcomer and the hot legs (HL) and compared the results to the break spectrum analysis in which such leakage was not modeled. The comparison showed that the PCT for the limiting break presented in FSAR Tier 2, Section 15.6.5 bounds the limiting break calculated for the break spectrum with downcomer to hot leg gap open. The limiting break for the DC/HL leakage case is the 17.8 cm (7.0 in.) break with a PCT of 820 °C (1,508 °F), versus 892 °C (1,638 °F) for the limiting case in FSAR Tier 2, Section 15.6.5. The comparison demonstrates that eliminating the downcomer to hot leg gap leakage is conservative. Therefore, the staff finds that omitting these flow paths is acceptable. Accordingly, the staff considers RAI 30, Question 15.06.05-24, resolved.

In RAI 30, Question 15.06.05-25, the staff requested the applicant to clarify the methodology for determining the various break locations (e.g., bottom, side, and top) in the RCP discharge leg. In a September 24, 2008, response to RAI 30, Question 15.06.05-25, the applicant stated that their SBLOCA methodology uses the Moody critical flow model to calculate break flow, as required by 10 CFR Part 50, Appendix K. As implemented, the Moody critical flow model uses a fixed slip ratio of 1.0, which yields homogeneous flow (equal phase velocities). As a consequence, this 10 CFR Part 50, Appendix K-required model calculates the same critical flow rate at the break regardless of break orientation. The response provides the requested clarification and notes that the break model is part of an approved methodology. The staff agrees that application of the Moody model will result in the same critical flow rate regardless of break orientation. Therefore, the staff finds the applicant's September 24, 2008, response to RAI 30, Question 15.06.05-25, acceptable, and considers RAI 30, Question 15.06.05-25, resolved.

In RAI 167, Question 15.06.05-40, the staff requested that the applicant provide copies of the pertinent input files for the limiting cases for both the LBLOCA and SBLOCA analyses. In a February 20, 2009, response to RAI 167, Question 15.06.05-40, the applicant submitted the requested files. These files were used as a reference in the staff's confirmatory calculations as described below.

Confirmatory Calculations

The staff has conducted several SBLOCA confirmatory calculations using both TRACE and RELAP5/MOD3.3. Best estimate break spectrum calculations were conducted with each code with only two SI trains available. The calculations used the same top peaked axial power profile used in the applicant's analyses. None of the TRACE or RELAP5 calculations predicted any significant fuel rod heatup. The calculations show that there is considerable conservatism in the applicant's analyses. Loop seal clearing was identified as a major difference between the staff's and the applicant's calculations. The applicant's simulations are biased (lowered) to retard their calculated loop seal clearing. No such bias was imposed in the staff's RELAP5 or TRACE calculations. Consequently, the staff's calculations generally predicted that more loops cleared. All loops predicted to clear tended to clear at the same time in the staff's simulations; whereas, in the applicant's simulations, there are generally two distinct periods of loop seal clearing, one at about the same time as the staff's calculations and another later, just prior to calculated PCT.

The staff added perturbations to its base calculations to help identify what assumptions were responsible for the large differences in PCTs between the staff's and applicant's analyses. A TRACE calculation was performed by the staff of the 16.5 cm (6.5 in.) break with two loop seals' clearing artificially delayed so they cleared at about the same time as in the applicant's calculation. With this perturbation, the TRACE calculation showed significant heatup with a PCT of 836 °C (1,537 °F). When this simulation was rerun using a decay heat multiplier of 1.2, the PCT increased to 937 °C (1,719 °F), a 100 °C (180 °F) increase. Similar perturbation cases were run with RELAP5. They showed a PCT increase of 122 °C (220 °F) when the decay heat multiplier was increased from 1.0 to 1.2. The staff's simulations indicate that there exists around 111 °C (200 °F) of conservatism in the applicant's simulations due to its decay heat assumption. Considerable additional conservatism is present in the applicant's calculations by virtue of its biasing the loop seals.

When conducting RELAP5 simulations, the staff noted that small changes in code input sometimes resulted in large changes in calculated PCT. The loop seal clearing process was identified as causing these large PCT variations. The calculation of loop seal clearing, or partial clearing, is very sensitive in RELAP5/MOD3.3, and small changes in clearing behavior have a large impact on calculated PCT. Since RELAP5 and S-RELAP5 have a shared pedigree, it is likely that the sensitivity of S-RELAP5 calculations to SG axial nodalization and the lack of a clear trend of calculated PCT with break size are due to differences in loop seal clearing.

The staff notes that following a SBLOCA, the MSRTs are automatically actuated on a safety injection signal. The MSRTs are programmed to reduce the steam generator pressure to 5.99 MPa (870 pounds per square inch absolute (psia)) at a rate corresponding to 82.2 °C (180 °F) per hour (/hr). The operators then continue the cooldown at a rate of 32.2 °C (90 °F) /hr. The staff also notes that during the system response following an SBLOCA, natural circulation can be interrupted and the system will enter a period of reflux-condenser cooling via the steam generators. This cooling mode involves counter-current flow in the primary system hot legs and steam generator tubes. In RAI 493, Question 15.06.05-113, the staff requested that the applicant provide the validation of S-RELAP to model counter-current flow under such conditions, and demonstrate the applicability of the data obtained at scaled facilities to the U.S. EPR. In addition, RAI 493, Question 15.06.05-113, requested that the applicant identify counter-current flow limits and associated effects on the U.S. EPR primary system heat extraction, demonstrate applicability of the S-RELAP5 counter-current flow limitation model to the U.S. EPR, including range coverage for mass flow rates, pipe diameter, inclination angle, and pressure, and specify the activation (flags) of the model in the U.S. EPR SBLOCA S-RELAP5 model. **RAI 493, Question 15.06.05-113, is being tracked as an open item.**

15.6.5.2.5 Combined License Information Items

There are no COL information items related to this area of review. The staff determined that no COL information items need to be included in FSAR Tier 2, Table 1.8-2, "U.S. EPR Combined License Information Items," for SBLOCA consideration.

15.6.5.2.6 Conclusions

The applicant has performed its SBLOCA analyses using an approved evaluation model (EM) that complies with the requirements of 10 CFR Part 50. The analyses incorporate a worst single failure (EDG failure) in combination with another EDG out for maintenance. The maximum calculated PCT is 892 °C (1,638 °F), which is below the 1,204 °C (2,200 °F) limit. Total calculated cladding oxidation at the PCT location was 0.38 percent, which is below the 17 percent limit. The calculated amount of hydrogen generated was 0.009 percent, which is below the one percent licensing limit. Independent staff best-estimate calculations showed no cladding heatup and, therefore, indicate that the applicant's calculated consequences of an SBLOCA are conservative. Since the calculations satisfy the licensing requirements, the staff finds the applicant's SBLOCA analyses acceptable with the exception of the open item identified above.

With exception of the open item identified above, the staff finds that the U.S. EPR design meets:

- 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 was met by assuring that initial conditions and SIS actuation setpoints are set to conservative values including consideration of measurement uncertainty.

- GDC 17, as it relates to onsite and offsite electric power systems, to ensure that SSCs function during normal operation, including AOOs is met in that the analysis was conservatively performed assuming loss of offsite power.

- GDC 35, as it relates to demonstrating that the ECCS would provide sufficient ECC to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts.

- 10 CFR Part 50, Appendix K, in that the SBLOCA analysis uses the methods and assumptions contained in the Appendix.

- 10 CFR 50.46, as it relates to ECCS equipment being provided that refills the vessel in a timely manner for an LOCA resulting from a spectrum of postulated piping breaks within the RCPB is met in that the analysis shows that the acceptance criteria are not exceeded.

- 10 CFR Part 100, as it relates to mitigating the radiological consequences of an accident is addressed in Section 15.0.3 of this report.

15.6.5.3 *Post-LOCA Long-Term Cooling*

15.6.5.3.1 Introduction

The FSAR provides long-term cooling analyses in accordance with 10 CFR 50.46, to confirm that: (1) The core remains cooled for the duration of the two-phase long-term cooling (LTC) phase; (2) the boron concentration in the core keeps the core subcritical; (3) boron precipitation will not obstruct core coolant flow; and (4) debris does not interrupt recirculation cooling.

15.6.5.3.2 Summary of Application

FSAR Tier 2, Section 6.3. "Emergency Core Cooling System," states that following postulated LOCAs, the SIS maintains fuel cladding temperature, cladding oxidation, hydrogen generation, core geometry, and long-term core temperature within the limits specified in 10 CFR 50.46. Long-term recirculation cooling is maintained by the LHSI function of the SIS. The SIS is supplied by water from the in-containment refueling water storage tank (IRWST) for long-term recirculation cooling following a LOCA with protection against loss of net positive suction head (NPSH) due to debris entrainment. An extra borating system (EBS) is also provided. The SIS injects automatically in response to the SI signal and does not credit operator intervention. The emergency coolant supply is enclosed within the containment, is constantly replenished by recirculated coolant flow, and provides a continuous supply of coolant for removal of decay heat during the SI phase. FSAR Tier 2, Section 6.3 states that the SIS is redundant and no single failure compromises the system safety functions. Vital power can be supplied from either the onsite or offsite power systems. FSAR Tier 2, Section 6.3 also states that the most limiting single active failure for the SIS, assumed to occur at the onset of the design basis LOCA event, is the complete loss of one SIS train. In addition, such analyses assume that another train of MHSI and LHSI is unavailable because of maintenance.

Long term heat removal is by boiling of the water in the reactor vessel. Internal vessel flow is by natural circulation. Steaming at the surface of the pool of water in the reactor vessel will result in concentration of the boric acid in the water. The U.S. EPR design employs hot-leg injection to control boric acid concentration in the core during pool boiling following a LOCA. FSAR Tier 2, Section 6.3 states that the LHSI system can be manually realigned during the accident recovery phase for hot-leg injection to both prevent boron precipitation and mitigate steaming from the break. To preserve the hot-leg injection function for mitigation of boron precipitation, there are four separate hot-leg connections, one for each of the SIS trains. FSAR Tier 2, Section 6.3.2.8, "Manual Actions," credits manual switchover to hot-leg injection at approximately 1 to 3 hours into the event to prevent boron precipitation in the event of a LBLOCA,

FSAR Tier 2, Section 15.6.5.4, "Long-Term Core Cooling," addresses several issues related to demonstrating adequate long-term cooling following a LOCA. These issues include boron precipitation, boron dilution during SBLOCA, and containment debris. FSAR Tier 2, Section 15.6.5.4.1, "Prevention of Boric Acid Precipitation," addresses the safety concern that boron in the coolant can concentrate and precipitate in the upper region of the core when there is protracted boiling following a LOCA. FSAR Tier 2, Section 15.6.5.4.2, "SBLOCA Boron Dilution," is related to Generic Safety Issue (GSI)-185, "Control of Recriticality Following

Small-Break LOCAs in PWRs," which raises a concern regarding the potential for recriticality during an SBLOCA if unborated water accumulates in the SGs and cold leg piping due to condensation and moves to the core as a slug. Generic Safety Issue (GSI)-191, "Experimental Studies of Loss-of-Coolant-Accident-Generated Debris Accumulation and Head Loss with Emphasis on the Effects of Calcium Silicate Insulation," related concerns regarding the potential damage to ECCS equipment and blockage of core channels due to debris in the water recirculated from the IRWST are addressed in FSAR Tier 2, Section 15.6.5.4.3, "IRWST Recirculation Cooling." Each of these issues is addressed by the staff in the following sections. In addition, the staff addressed, in a separate section below, the safety concern related to the potential for core two-phase mixture level suppression and associated core uncovery during the post-LOCA long-term core cooling phase.

15.6.5.3.3 Regulatory Bases

The relevant requirements of NRC regulations for this area of review, and the associated acceptance criteria, are given in NUREG-0800, Section 15.6.5 and are summarized below. Review interfaces with other SRP sections also can be found in NUREG-0800, Section 15.6.5.

1. GDC 13, "Instrumentation and Control," 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 35, "Emergency Core Cooling," as it relates to demonstrating that the ECCS would provide abundant ECC to satisfy the ECCS safety function of transferring heat from the reactor core following any loss of reactor coolant at a rate that (1) fuel and clad damage that could interfere with continued effective core cooling would be prevented, and (2) clad metal-water reaction would be limited to negligible amounts.

3. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," as it relates to ECCS equipment being provided that refills the vessel in a timely manner for a LOCA resulting from a spectrum of postulated piping breaks within the RCPB.

4. 10 CFR 50.46, as it relates to Paragraph (b)(5), "Long-term cooling." After any calculated successful initial operation of the 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.

Acceptance criteria adequate to meet the above requirements include:

1. An evaluation of ECCS performance has been performed by the applicant in accordance with an evaluation model that satisfies the requirements of 10 CFR 50.46.

2. The analyses must demonstrate sufficient redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities such that the safety functions could be accomplished assuming a single failure in conjunction with the availability of onsite power (assuming offsite electric power is not available, with onsite electric power available; or assuming onsite electric power is not available with offsite electric power available). Additionally the LOCA methodology used and the LOCA analyses should be shown to apply to the individual plant by satisfying

10 CFR 50.46(c)(2), and the analysis results should meet the performance criteria in 10 CFR 50.46(b).

Additional guidance is provided in:

1. RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident,” Revision 3, November 2003

2. GL 2004-02, “Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,” NRC, September 2004

3. Bulletin No. BL 93-02, “Debris Plugging of Emergency Core Cooling Suction Strainers,” May 11, 1993

4. NUREG-0933, Section 2, Issue 185, “Control of Recriticality Following Small-Break LOCAs in PWRs”

5. NUREG -0933, Section 3, Issue 191, “Assessment of Debris Accumulation on PWR Sump Performance,” (Revision 2)

15.6.5.3.4 Technical Evaluation

15.6.5.3.4.1 *Long-Term Cooling Core Mixture Level*

FSAR Tier 2, Section 15.6.5.4 does not explicitly demonstrate that the core remains covered by two-phase coolant mixture during the post-LOCA long-term cooling phase. Reformation of the loop seals due to ECCS injection during the U.S. EPR post-LOCA long-term cooling phase can cause suppression of the two-phase mixture level on the reactor core side. If the level drops below the top of the active fuel, cladding heat-up and oxidation can occur. In RAI 241, Question 15.06.05-51, the staff requested that the applicant provide the results of a thermal hydraulic analysis demonstrating that the core remains consistently covered by a two-phase coolant mixture during the post-LOCA long-term cooling phase which can, without intervention, in principle last for months. In a November 25, 2009, response to RAI 241, Question 15.06.05-51, the applicant stated that a quasi-steady static-balance analysis approach is used to calculate the core two-phase mixture level as a function of decay power during the post-reflood period. The static balance approach shows that the minimum core collapsed liquid level is the same as the elevation at the top of the cross-over pipe U-Bend (loop seal), while the two-phase mixture level rises above the collapsed liquid level because of boiling and void generation in the core. The results from computations using three different two-phase flow correlations and presented in the applicant’s November 25, 2009, response to RAI 241, Question 15.06.05-51, predict that the top of the active fuel (TAF) remains covered by a two-phase mixture over the entire post-reflood long-term cooling period of interest assuming nominal input parameters. Essential to this conclusion is that the elevation at the top of the cross-over piping U-bend (loop seal) is only 3.048 cm (0.1 ft) below the TAF.

Key assumptions in the applicant’s original analysis in the November 25, 2009, response to RAI 241, Question 15.06.05-51 are:

- LHSI is operational and injects sufficient water to offset the loss through the largest break. LHSI capacity for a single pump at 99.97 kPa (14.5 psia) is 151.5 kg/sec

(334 lb/sec), which provides makeup water to the core where the steaming rate is less than 5.54 kg/sec (10 lb/sec) in the post reflood period of interest.

- When the LHSI system is switched to hot leg injection, it will deliver more than 75 percent of its subcooled water to the hot leg, where it can condense steam and also deliver a large quantity of water directly to the top of the core; however, hot leg injection is not credited in this analysis, which is a conservative assumption.

- A cold leg break between the RCP and reactor vessel is the most limiting location because of restricted steam venting though the loop seal. Other break locations do not impose significant limitations to the venting of steam and do not cause a depression of the collapsed liquid level in the core.

- The analysis is independent of break size and geometry because of the conservative assumption that LHSI maintains postulated water levels in the RCS with the LPSI flow to the reactor equal to the steaming rate and the remainder of the flow spilling rather than refilling the RCS.

- Decay heat in the core generates steam. Subcooled LHSI water enters the downcomer and flows to the core to replenish the water that has evaporated to steam.

- The water and steam are saturated at the selected primary system pressure except for the water in the downcomer and core inlet. The subcooled water temperature at the core inlet is equivalent to the LHSI temperature. Subcooled water at the core inlet produces a lower two-phase mixture level than if saturated water is assumed since part of the decay heat is consumed to bring the subcooled liquid to the saturation temperature. This reduces the steam generation in the core, which results in reduced voiding and a suppressed two-phase mixture level.

The calculations show that using nominal bypass flow resistance and atmospheric pressure, the minimum collapsed liquid level occurs beyond the time period of the computation as seen from Figure 15.06.05-51-2 in the November 25, 2009, response to RAI 241, Question 15.06.05-51. Increasing the pressure to about 202.7 kPa (29.4 psia) increases the steam density and bypass flow rate. The minimum collapsed liquid level in this case is reached earlier in the transient. Decreasing the bypass flow resistance coefficient in the 101.3 kPa (14.7 psia) case increases the bypass flow further and the minimum is reached even earlier, see Figure 15.06.05-51-4 in the November 25, 2009, response to RAI 241, Question 15.06.05-51. In all cases, and for all three two-phase flow correlations, the two-phase mixture level is always above the TAF in the applicant's analysis.

As the applicant discovered an error in this original response to RAI 241, Question 15.06.05-51, a revised response to this question was submitted on May 25, 2010. The revised response to RAI 241, Question 15.06.05-51, in addition to adding technical detail, was concerned with two main items as described below.

- The Wilson and Cunningham-Yeh void fraction correlation coefficients were corrected to agree with their original References 3 and 4 cited in the November 25, 2009, response to RAI 241, Question 15.06.05-51, rather than using those reported in References 1 and 2.

- The analytical solution was revised to limit vapor generation to the active core. Vapor generation was set to zero above the TAF.

In a November 25, 2009, response to RAI 241, Question 15.06.05-51, the applicant explained that addressing both of the above described items changed the details presented in the original response; however, the overall finding that the core remained covered by a two-phase mixture remained unchanged in the May 25, 2010, response to RAI 241, Question 15.06.05-51.

Based on the evaluation of the May 25, 2010, response to RAI 241, Question 15.06.05-51, and an audit of the computational model, which took place at the AREVA local office in Rockville, MD on June 23, 2010, new issues emerged concerning the calculational model. These items, formulated as specific follow-up questions to RAI 241, Question 15.06.05-51, or as additionally identified items, were further discussed with the applicant at a public meeting on July 15, 2010. In RAI 403, Questions 15.06.05-69 through 15.06.05-73 and 15.06.05-75 through 15.06.05-77, the staff issued these questions as described in detail below.

In RAI 403, Question 15.06.05-69, the staff requested that the applicant substantiate the applicability and accuracy of the two-phase level swell correlations applied in the November 25, 2009, response to RAI 241, Question 15.06.05-51, under post LOCA low pressure conditions. In a May 31, 2011, response to RAI 403, Question 15.06.05-69, the applicant provided the technical justification for the use of the Cunningham-Yeh two-phase correlation in the level swell analysis. The staff was concerned that the correlation had not been validated for low pressure. The applicant explained that the correlation had been validated with the Hitachi electrically heated, full length, 7x7 rod-bundle, level swell experiments, which were conducted at atmospheric pressure. The staff has reviewed the information provided in the May 31, 2011, response to RAI 403, Question 15.06.05-69, and finds it acceptable, since the requested validation has confirmed the applicability of the correlation at atmospheric pressure. Accordingly, the staff considers RAI 403, Question 15.06.05-69, resolved.

In RAI 403, Question 15.06.05-70, the staff requested that the applicant explain why crediting bypass, identified as Path 1 in Figure 15.06.05-51-1 in the revised response to RAI 241, Question 15.06.05-51, was appropriate and conservative and justify the minimum gap resistance value used in the analysis. **RAI 403, Question 15.06.05-70, is being tracked as an open item.**

In RAI 403, Question 15.06.05-71, the staff requested that the applicant show quantitatively the sensitivity of the two-phase mixture level to a top skewed axial power distribution under the most limiting physically realizable top peak profile. In a May 31, 2011, response to RAI 403, Question 15.06.05-71, the applicant provided the technical justification for a uniform axial power distribution. The staff believed that a top skewed power distribution would result in a lower two-phase mixture level due to less void swell. The applicant performed quantitative analyses to show that the two-phase mixture level was insensitive to the axial power shape. The staff has reviewed the draft response and finds it acceptable, since the requested information shows the insensitivity of the collapsed mixture level and justifies use of a bottom peaked shape consistent with the boron precipitation analyses. Accordingly, the staff considers RAI 403, Question 15.06.05-71, resolved.

In RAI 403, Question 15.06.05-72, the staff requested that the applicant explain the sensitivity of the two-phase mixture level in the core to loop seal water temperature. In a May 31, 2011, response to RAI 403, Question 15.06.05-72, the applicant provided the technical justification for use of an assumed saturated downcomer and subcooled cold leg and loop seal in their model.

The staff was concerned about the sensitivity of the two-phase mixture level in the core to loop seal water temperature. Sensitivity studies performed by the applicant examined the impact of assumptions with regard to the liquid temperature both in the cold leg piping and downcomer on the predicted collapsed and two-phase mixture levels in the core region. The reported results showed that the impact of the water temperature in the cold leg and loop seal on the collapsed water level was small. For the range of considered temperature assumptions used to determine the liquid density in the loop seal and downcomer, the core was always well covered by two-phase mixture. The predictions showed that the case with saturated downcomer and subcooled cold leg and loop seal liquid was the most conservative as it produced the lowest collapsed and two-phase mixture levels. The May 31, 2011, response to RAI 403, Question 15.06.05-72, stated that the conditions for this limiting case are used in all calculations unless stated otherwise. The staff has reviewed the response and finds it acceptable, since the assumptions from the limiting case with saturated liquid in the downcomer and subcooled liquid in the cold leg piping that conservatively minimize the predicted core levels are used in the analyses. Accordingly, the staff considers RAI 403, Question 15.06.05-72, resolved.

In RAI 403, Question 15.06.05-73, the staff requested that the applicant provide a conservative estimate of the degree of depression of the liquid level in the horizontal section of the loop seal pipe to provide a steam relief path. In a May 31, 2011, response to RAI 403, Question 15.06.05-73, which is a follow-up question to RAI 241, Question 15.06.05-51, the applicant provided technical justification that depressing the water level in the loop seal to the bottom of the pipe following a LOCA has a small impact on the collapsed water level. The staff had questioned what degree of depression of the liquid level below the top of the loop seal horizontal piping was needed to vent steam through the loop seal to prevent further pressure buildup and corresponding level suppression in the core. The applicant showed that there is not a one-to-one relationship between the water level in the loop seal and the collapsed level when the void fraction in the loop seal vertical leg is high. The staff has reviewed the response and finds it acceptable, since the analytic results show that the collapsed liquid level is almost insensitive to the water level in the loop seal. Accordingly, the staff considers RAI 403, Question 15.06.05-73, resolved.

In RAI 403, Question 15.06.05-75, the staff requested that the applicant provide the results with the break and liquid level located at the top elevation of the discharge piping. In a May 31, 2011, response to RAI 403, Question 15.06.05-75, the applicant provided analyses that showed that the collapsed liquid level was insensitive to the location of the cold leg pipe break. The staff had been concerned that the assumed location of the pipe break was not limiting and requested results with the break and liquid level located at the top elevation of the discharge piping. The applicant also confirmed that the highest elevation of the SI lines downstream of the check valve is at the side injection nozzle at the cold leg discharge piping, and is at the cold leg centerline. The analyses address the staff's concerns in that the applicant showed that the collapsed liquid level was insensitive to the break location. Accordingly, the staff considers RAI 403, Question 15.06.05-75, resolved.

In RAI 403, Question 15.06.05-76, the staff requested that the applicant describe the model for calculating the void fraction in the loop seal vertical section. In a May 31, 2011, response to RAI 403, Question 15.06.05-76, the applicant stated that the drift-flux model was used to compute the void fraction in the vertical leg of the loop seal under the pump. The drift-flux model is a widely used model to compute two-phase flow void fraction. The applicant explained that the model implementation was intended to minimize the void fraction. In addition, the applicant stated that the shallow loop seal design in the U.S. EPR limits the impact of the implemented modeling choices. The staff finds the response acceptable as the applicant identified how the void fraction had been calculated and explained how the model was implemented to minimize void fraction. The staff agrees that reduced void fraction in the loop seal vertical section impacts the predictions in a conservative manner as it increases the two-phase level suppression in the core. Accordingly, the staff considers RAI 403, Question15.06.05-76, resolved.

In RAI 403, Question 15.06.05-77, the staff requested that the applicant provide an analytical description of the model used to compute the system response for the two-phase mixture level during the post-LOCA long-term cooling phase. In addition, the staff requested a copy of the computer program used to produce the analysis and plots of predictions for key parameters. In a May 31, 2011, response to RAI 403, Question 15.06.05-77, the applicant provided plots of void distribution, core steaming rate, vapor mass flow rates, liquid density and void fraction in the loop seal, and mass flow rate through the bypass. The applicant also provided input parameters and selected computer output for two cases that were analyzed along with plots of predicted parameters for each case. The plots depicted axial distributions for the core void fraction, core linear power, and vapor volumetric flux. In addition, the core steaming rate, flow rate through the loops, and flow rate through the bypass were plotted as a function of time. As part of the response, the applicant also provided a copy of the computer program used to produce these results. The staff had requested this information so that it would be available to staff to support the review of the model and confirm the applicant's analysis. Except for the analytical description of the model, the response answers the staff's data request.

In the May 31, 2011, response to RAI 403, Question 15.06.05-77, the applicant reiterated that:

> Long term core cooling analysis is performed with a revised and expanded analytical model to show that the U.S. EPR core remains covered by a two-phase mixture that cools the fuel rods long term. This result is due to the shallow U.S. EPR loop seal design. The elevation at the top of the cross-over piping U-bend (loop seal) is 30 mm below the top of the active fuel, and only a small level swell is needed to cover the active fuel with a two-phase mixture.

The staff notes that the favorable geometry of the U.S. EPR, and specifically the shallow loop seal, results in a robust conclusion in that a simple pressure balance with the shallow loop seal shows that the core will remain covered with a two-phase mixture level. The applicant's analysis has not taken credit for switchover to hot leg injection. This is another robust conservatism in the approach by the applicant.

As stated in the May 31, 2011, response to RAI 403, Question 15.06.05-77, the computer program implementing a quasi-steady static-balance model for calculating the U.S. EPR core two-phase mixture level during the post-reflood period was provided to NRC by letter dated November 19, 2010. Results obtained with this code or its routines were used in the responses to RAI 403, Questions 15.06.05-69 through 15.06.05-73 and 15.06.05-75 through 15.06.05-77, considered above, as well as in the May 31, 2011, response to RAI 403, Questions 15.06.05-64 and 15.06.05-65, on boron precipitation reviewed below. However, as described above, RAI 403, Question 15.06.05-70, which involves this program, is still open and the program is still under review by the staff. **RAI 403, Question 15.06.05-70, is being tracked as an open item.**

15.6.5.3.4.2 *Boron Dilution and Return to Criticality Following a LOCA*

This section presents safety aspects of the review of the FSAR Tier 2, Chapter 15, "Transient and Accident Analyses," pertinent to GSI-185, "Control of Recriticality Following Small-Break LOCAs in PWRs." GSI-185 addresses concerns related to potential conditions arising during an SBLOCA scenario when a slug of de-borated water can accumulate in cold leg loop seal piping due to steam condensation in the SG during reflux cooling. Upon reestablishment of the natural circulation in the corresponding primary reactor coolant loop, the de-borated water volume is transported into the reactor vessel and core, potentially causing a return to criticality and fuel damage.

FSAR Tier 2, Section 15.6.5.4.2 refers to tests performed by the applicant at the PrimärKreisLauf (PKL) integral-loop test facility to address boron dilution concerns during SBLOCA. This FSAR section suggests that the tests demonstrated that the restart of natural circulation was preceded by a period of intermittent circulation, with the circulation starting first in one active loop followed independently by the remaining active loops.

The staff also reviewed pertinent portions of AREVA Technical Report ANP-10288P, Revision 0, "U.S. EPR Post LOCA Boron Precipitation and Boron Dilution," November 2007, and determined that it was necessary to review additional technical documents and information related to the assessment of the formation and volume of the diluted slug during the reflux condenser mode, its transportation to the core inlet via the cold leg and downcomer along with associated coolant mixing. During an audit activity, which took place at the AREVA local office in Rockville, MD on December 3-5, 2008, such additional technical information was provided and examined by the staff. The staff reviewed documentation related to boron dilution and boron precipitation effects for the U.S. EPR design, including reports that summarized results from S-RELAP5 LOCA analyses with various break sizes. The staff reviewed the assumptions used in the applicant's detailed calculation. Following the audit, the staff issued RAI 167, Questions 15.06.05-29 through 15.06.05-34, to obtain the relevant information relating to the audited documentation. The staff evaluation of these RAIs is discussed below.

The above issues pertain to how much de-borated water might be generated and where it would be collected. The model predicts how the de-borated water might be transported to the core and the criticality consequences of such an event. The applicant provided FANP NGTT1/04/en/04, Revision A, "Final Report of the PKL Experimental Program within the OECD/SETH Project," Framatome ANP, December 2004 for audit. The PKL facility is a full height but narrow test facility, and the applicability of the qualitative observations at PKL to the U.S. EPR is not straight forward. This report describes incoherent resumption of natural circulation following an SBLOCA. As described below, in RAI 403, Question 15.06.05-63, the staff requested that the applicant provide relevant information from this report to be submitted on the docket.

Based on the then-current status of the review of FSAR Tier 2, Section 15.6.5.4.2, AREVA Technical Report ANP-10288P, Revision 0, the associated analyses performed by the applicant, and the findings from the audit on December 3-5, 2008, the staff determined that additional information related to SBLOCA boron dilution was needed before making a conclusion regarding the acceptability of the U.S. EPR design in response to a boron dilution return to criticality event following an SBLOCA. Neither AREVA Technical Report ANP-10288P, "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution," Revision 0, November 2007, nor other documentation available and considered at the audit, addressed areas related to criticality consequences of the postulated event. To assess the safety aspects related to criticality consequences if a slug of unborated water were to be transported to the reactor vessel inlet by resumption of natural circulation, the staff raised issues pertaining to how much de-borated water might be generated and where it would be collected. In summary, the model predicts how the de-borated water might be transported to the core and the criticality consequences of such an event.

Areas in which the staff sought additional information were formulated as questions in RAI 167, submitted to the applicant on January 23, 2009, as part of the overall review of the U.S. EPR design features and performance to address GSI-185. The question on SBLOCA boron dilution, issued as part of RAI 167, included Questions 15.06.05-29 through 15.06.05-34, Question 15.06.05-36, Question 15.06.05-37, and Question 15.06.05-42. The applicant provided the response to RAI 167, Question 15.06.05-42, on March 31, 2009. The responses to RAI 167, Questions 15.06.05-30 through 15.06.05-33 and Question 15.06.05-36, were provided on April 30, 2009. The responses to the remaining questions, RAI 167, Questions 15.06.05-29, 15.06.05-34, and Question 15.06.05-37, were provided on May 28, 2009. The staff evaluation of the responses to these RAI 167 questions is provided below.

During the course of the review process, in January 2010, the applicant issued AREVA Technical Report ANP-10288P, "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report," Revision 1, which was also reviewed by the staff.

In RAI 167, Question 15.06.05-29, the staff requested that the applicant describe possible accumulation of condensate in the SG exit chamber and the attached cold leg nozzle/piping or consider the liquid conditions in the RCP discharge pipe. In addition, the staff requested that the applicant provide an assessment of the total amount of condensate generated in each loop.

In a May 28, 2009, response to RAI 167, Question 15.06.05-29, the applicant identified five conditions necessary for accumulation of de-borated coolant in the primary circuit during an SBLOCA accident and stated that the mass and boron concentration of the de-borated slug depended on several factors, including the amount of time spent in the reflux-condensation mode, the boiling and condensation rates, the characteristics and the behavior of the refill, and the system geometry. The staff concurs with this description. The applicant's response includes Figure 15.6.5-29-1, and presents a sketch of the U.S. EPR crossover piping section illustrating the volume for possible accumulation of de-borated fluid between the SG exit chamber and the RCP suction and the volume of liquid from the pump suction to the pump discharge. This slug accumulation geometrical volume was reported as 4.84 m^3 (171 ft^3). To account for any additional small amounts of possible de-borated fluid accumulation in the bottom of the pump casing or SG outlet plenum, this volume was increased by 25 percent and reported as being equal to 6.12 m^3 (216 ft^3) as an assumption to be used in the analysis. As described below, the staff asked further questions regarding this topic.

In a June 1, 2009, response to RAI 167, Question 15.06.05-29, the applicant did not provide an assessment of the amount of condensate generated by reflux condensation as was requested. Instead, predictions for the liquid content in control areas defined as "SG outlet plenum," "loop crossover pipe," and "cold leg" were presented. The staff was concerned that additional deborated water could accumulate in the steam generator lower plenum in excess of the 6.12 m (216 ft^3) and requested that the applicant provide additional information. In the May 28, 2009, response to RAI 167, Question 15.06.05-29, the applicant presented results from the SBLOCA analyses that showed transient predictions for the mass of liquid present in three different control volumes in each of the four primary loops: (1) SG outlet plenum; (2) loop crossover pipe; and (3) cold leg. The results were presented for the 3.81 cm (1.5 in.), 5.08 cm (2 in.), 7.62 cm (3 in.), and 10.16 cm (4 in.) break size cases.

The applicant pointed out that in the three S-RELAP5 cases for which natural circulation ceased, and for which the break size ranged from 5.08 cm (2 in.) to 10.16 cm (4 in.), the crossover pipe was never predicted to be completely filled prior to system refill so that the slug of water collected during the boiler-condenser phase was less than the crossover pipe volume. In addition, it was stated that during the refill stage, the crossover pipe and SG outlet plenum were predicted to be refilled prior to the restart of natural circulation. Therefore, AREVA claimed that this process added boron to the original slug. AREVA also explained that the refill process occurred largely due to backflow predicted through the RCS pumps. With regard to the 5.08 cm (2 in.) break case, the applicant stated that for loops without MHSI flow, such backflow occurred when the RCS pressure was predicted to drop below the LHSI shutoff head pressure and SI increased, thus refilling the system before the first restart of natural circulation in such loops.

Furthermore, the applicant stated that such a backflow process provided boron to the original slug in the crossover pipes in affected loops before these pipes would experience an erratic flow and establishment of continuous forward natural circulation.

Based on information the staff obtained at an audit at AREVA headquarters in Lynchburg, VA on March 24, 2010, and March 25, 2010, the staff formulated a follow-up question. Following a discussion at a public meeting on July 15, 2010, the question was issued as RAI 403, Question 15.06.05-61, on June 2, 2010. In RAI 403, Question 15.06.05-61, the staff requested that the applicant provide a calculation of the amount of condensate generated by reflux condensation accounting for conditions with EFW supply present for two or more SGs. The mass of steam due to decay heat is considerably larger than the volume of the loop seals. The staff requested that the applicant provide a mass balance accounting for the steam generated in the core by decay heat, steam lost out the break, steam condensed in the SGs, condensate in countercurrent flow returned to the reactor vessel hot leg, condensate collected on the SG, loop crossover pipe, and cold leg, condensate lost out the break, and condensate transported to the vessel downcomer. The staff also requested that the applicant provide a description of situations involving stratified conditions in the cold leg and downcomer regions. **RAI 403, Question 15.06.05-61, is being tracked as an open item.**

In RAI 167, Question 15.06.05-30, the staff requested that the applicant provide an explanation of the availability, quantity, and distribution of the ECCS flow to each individual loop consistent with the analysis assumptions made in the applicant's SBLOCA analysis. The staff requested that the applicant consider, in particular, the possibility for conditions that would deprive a loop from any ECCS injection and, if such a possibility existed, address associated effects on the restart of the natural circulation and mixing in the corresponding cold loop.

In an April 30, 2009, response to RAI 167, Question 15.06.05-30, the applicant answered the question by systematically describing the liquid volume and boron concentration for the sources of ECCS water (IRWST, accumulator tanks, EBS tanks) as shown in table 15.06.05-30-1 of the April 30, 2009, response to RAI 167, Question 15.06.05-30), and the arrangement of the MHSI system, LHSI system, and the EBS system. In addition, multiple combinations of ECCS injection capacity associated with single failure and preventive maintenance assumptions were identified for each individual loop taking into consideration: (1) The LHSI cross-connects; (2) an assumed single failure; and (3) an assumed ECCS train out of service for preventive maintenance. As a result, a spectrum of seven injection scenarios were identified as shown in Table 15.06.05-30-2 of the April 30, 2009, response to RAI 167, Question 15.06.05-30. In addition, the ECCS and EBS functional arrangement, shown in Table 15.06.05-30-2, is illustrated in Figure 15.06.05-30-1 of the April 30, 2009, response to RAI 167, Question 15.06.05-30. The applicant also stated that FSAR Tier 2, Section 15.6.5 SBLOCA analyses were synonymous with Scenario 7, assuming that SI Trains 2 and 3 were inoperable with one failed and the other out of service for preventive maintenance. Accordingly, the analyses also assumed the failure and maintenance outage of the SG EFW in the same loops. Thus, only Loops 1 and 4 are modeled as having active MHSI and LHSI pumps. The applicant identified that there were, within the design basis, scenarios for which one or two loops would not receive SI flow. The applicant argued that these loops would receive backfill (through the idle RCP) from the reactor vessel with water at the downcomer boron concentration as the plant refills. With respect to one such scenario, Scenario 6, in which two loops are modeled as completely deprived of SI flow if the EBS is not manually activated, the applicant stated that for a selected break size, the RCS would be refilled with essentially the same timing and sequence as in Scenario 7, since the quantity of water going into the RCS is independent of the SI combinations. Furthermore, in the April 30, 2009, response to RAI 167, Question 15.06.05-30, the applicant stated that during refill of the RCS crossover pipe, there would be backflow of borated water from the cold legs receiving SI to the RCS loops without SI. The applicant stated that these backflows and the dynamics of the refill would re-borate those crossover pipes before two-phase natural circulation begins. Therefore, the applicant concluded that the analysis provided a reasonable representation of the system behavior for all injection scenarios. The applicant also stated that the EBS system was not modeled in the S-RELAP5 analysis of these events, because the EBS flow rates were small in comparison to the SI flow rates, and the presence or absence of the EBS was, therefore, negligible with regard to the time of the restart of natural circulation.

The staff determined that it is not self evident that the dynamics of refill will reborate the cross over pipe. If the cold leg pipe break is below the elevation of the impeller discharge, there does not appear to be a mechanism for backfill of the crossover pipe. Similarly, it is not apparent that Scenario 6 is the worst case.

Based on information the staff obtained at an audit at AREVA headquarters in Lynchburg, VA, on March 24 and 25, 2010, and following a discussion at a public meeting on July 15, 2010, in RAI 403, Question 15.06.05-62, the staff requested that the applicant demonstrate that de-borated condensate accumulated in one or more loops (SG plena, loop seals, cold legs, downcomer) that are completely or partially deprived of SI will not pose a recriticality threat to the U.S. EPR if transported towards the core due to natural circulation restart. Specifically, the staff requested that the applicant identify and consider limiting conditions, assumptions, and scenarios in terms of condensate accumulation in individual loops and associated regions as well as transportation mechanisms involving possible restart in multiple primary loops. The staff also requested that the applicant explain which of the postulated scenarios identified in the April 30, 2009, response to RAI 167, Question 15.06.05-30, provides the greatest challenge to core re-criticality. **RAI 403, Question 15.06.05-62, is being tracked as an open item.**

In RAI 167, Question 15.06.05-31, the staff requested that the applicant provide a description of the SG secondary response in terms of pressure, temperature, and liquid inventory conditions for each individual SG consistent with the analysis assumptions made, including EFW availability and depressurization considerations. SG secondary conditions determine the SG heat extraction capability from the primary circuit, which is the driving mechanism for condensate generation in the U-tube bundle in a reflux-condenser mode. In an April 30, 2009, response to RAI 167, Question 15.06.05-31, the applicant presented graphical results for the SG secondary-side pressures, mass inventories, and EFW mass flow rates for each individual SG for the SBLOCA cases with 3.81 cm (1.5 in.), 5.08 cm (2 in.), 7.62 cm (3 in.), and 10.16 cm (4 in.) break sizes analyzed. The results for the 15.24 cm (6 in.) break were not included as the SGs were not predicted to depressurize below the RCS pressure until approximately 12,000 sec into the event. In all cases analyzed, it was assumed that the SGs in Loops 2 and 3 did not receive EFW flow, since the single failure criterion was applied to one train, and the other train was assumed to be out of service for preventive maintenance. The staff considers the explanation acceptable, since the graphs show quantitative results describing the SG secondary conditions during the cool down process, which is the driving mechanism for generation of condensate in the primary RCS components. Therefore, the staff considers RAI 167, Question 15.06.05-31, to be resolved.

In RAI 167, Question 15.06.05-33, the staff requested that the applicant clarify the adequacy of the S-RELAP model used to analyze the boron dilution and boron precipitation effects in the U.S. EPR SBLOCA analysis report. In an April 30, 2009, response to RAI 167, Question 15.06.05-33, the applicant stated that the S-RELAP5 model was assessed against the tests in the PKL Test Facility, which were designed to replicate system refill and the restart of natural circulation, and referred to the discussion in the response to RAI 167, Question 15.06.05-36. The applicant stated that the S-RELAP5 analyses followed qualitatively the same evolution of the event as the PKL tests with the flow dynamics being very similar to that observed in the PKL tests after the boiler condenser stage, including the backflow behavior during refill, which was found to increase the boron concentration in the crossover pipe prior to the restart of natural circulation. The applicant stated that the S-RELAP5 code was not used to track boron concentration. Instead, the liquid volumes and relevant flow rates were used in a follow-on calculation of the boron concentrations to establish the boron concentration in the liquid sent to the reactor vessel and subsequently to the core upon restart of natural circulation.

Based on information obtained at an audit on March 24 and 25, 2010, in RAI 167, Question 15.06.05-36, the staff requested that the applicant provide additional information as discussed below.

In RAI 167, Question 15.06.05-34, the staff requested that the applicant clarify the criticality consequences if a coherent slug of unborated water of the quantity corresponding to the volume of a single crossover pipe were to be transported to the reactor vessel inlet by resumption of natural circulation. In a May 28, 2009, response to RAI 167, Question 15.06.05-34, the applicant presented the results of additional neutronics analyses evaluating the criticality effect caused by a slug of de-borated water entering the reactor core. The volume of the slug was varied assuming it was equal to 25 percent, 50 percent, 75 percent, and 100 percent of the maximum de-borated fluid volume accumulated in a single crossover pipe, which was estimated conservatively at 6.12 m^3 (216 ft^3) in the April 30, 2009, response to RAI 167, Question 15.06.05-29. Thus, the minimum slug volume was set at 1.53 m^3m3 (54 ft^3), or one quarter of 6.12 m^3 (216 ft^3), and the maximum one was equal to 6.12 m^3 (216 ft^3). In addition, the limiting case with respect to the maximum possible slug volume was considered assuming a de-borated slug that would occupy the entire core. In this case, the slug volume was set equal to 24.8 m^3 (877.1 ft^3). The results from the analysis were reported as concentrations, given in Table 15.6.5-34-1, in units of parts per million (ppm) of natural boron in the de-borated slug of each volume size sufficient to maintain a core shutdown margin of one percent. The applicant also states that the results included an additional 100 ppm for conservatism. Thus, in the case of a slug volume of 6.12 m^3 (216 ft^3), the analysis determined that a concentration of 1,396 ppm as natural boron was needed to maintain the predetermined shutdown margin of one percent. In the limiting case when the slug occupied the entire core, the analysis revealed that a higher concentration value of 1,519 ppm as natural boron was necessary to maintain the same core shutdown margin. The staff considered that the results, although derived with some conservative assumptions, involved a "flat" slug shape assuming that the slug was spread uniformly across the entire core flow area. The staff determined the assumption of "flat" slug shape non-conservative from a neutron balance standpoint. The only assessment result free of this non-conservatism was obtained for the limiting case in which the slug occupied the entire core with a corresponding volume of 24.8 m^3 (877.1 ft^3). For this case, the limiting boron concentration was determined as 1,519 ppm of natural boron, and it is this bounding value that is used in the staff's overall conclusion related to boron dilution. Therefore, the staff considers RAI 167, Question 15.06.05-34, resolved.

In RAI 167, Question 15.06.05-35, the staff requested that the applicant show that boron addition from all fluid sources such as the accumulators and IRWST is sufficient to ensure that following an LBLOCA, the reactor will be shutdown at cold conditions. In a February 19, 2009, response to RAI 167, Question 15.06.05-35, the applicant stated that neglecting the benefits of concentration of boron in the mixing volume of the core following an LBLOCA, and without crediting the negative reactivity worth of xenon-135, the resulting minimum IRWST boron concentration is sufficient to achieve at least a one percent shut down margin (SDM) at cold conditions. The analyses were performed at the BOC of 18-month and 24-month fuel cycles, which would be the most reactive cold conditions. All control rods are assumed to be stuck out of the core, which is an additional conservatism. Credit is taken for the capability of the 37 percent enriched boron-10, which is used throughout the EPR, to absorb neutrons. Injection from the safety-related emergency borating system increases the margin. In view of the foregoing, the staff finds that the analysis was conservatively performed, shows adequate margin, and therefore is acceptable. Accordingly, the staff considers RAI 167, Question 15.06.05-35, resolved.

In RAI 167, Question 15.06.05-36, the staff requested that the applicant explain why the qualitative behavior observed in the PKL tests is applicable to the U.S. EPR. This question referred to PKL tests performed at AREVA's PKL test facility in Erlangen, Germany, to evaluate boron dilution effects in PWRs and referenced by the applicant in AREVA Technical Report ANP-10288P, "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution," Revision 0, November 2007.

In an April 30, 2009, response to RAI 167, Question 15.06.05-36, the applicant identified the qualitative behaviors demonstrated by the PKL tests that were used as part of the justification for the analytical methodology for evaluating the U.S. EPR boron dilution event. The applicant made the following experimental observations: (1) The backfill of the SG outlet plenum and downside tubes via the cold legs occurs prior to the establishment of erratic or sustained natural circulation; (2) the experiments showed the initiation of natural circulation in more than one loop at a time; and (3) the experiments showed an extended period of erratic circulation prior to sustained liquid natural circulation during system refill. The applicant stated that the evidence of flow regimes or conditions, such as erratic two-phase flow before initiation of natural circulation, was credible as a backup to full scale plant simulations.

The Organization for European Community Development (OECD) sponsored Primärkreislauf-Versuchsanlage (PKL) III F test program was planned as comprising four test groups of which group F1 was dedicated to investigations of boron dilution due to natural forces and restart of natural circulation after refilling of the primary system during SBLOCAs. Test group F1 included four different tests, F1.1 through F1.4, with Test F1.1 assuming injection of emergency core coolant by the SI pumps into all four cold legs, Test F1.2 investigating accumulation of low borated water as a function of the primary side inventory, Test F1.3 assuming injection of emergency core coolant by the SI pumps into two out of four hot legs, and Test F1.4 simulating injection of emergency core coolant by the SI pumps into two out of four cold legs. In its response, the applicant described briefly the conditions only for Test F1.1 and compared those to the U.S. EPR S-RELAP5 SBLOCA analysis assumptions without quantitative analysis except for specifying the SG cooldown rate (56 °C/hr=100.8°F/hr in the test versus 50 °C/hr=90°F/hr for the U.S. EPR), the test break size (5.46 cm (2.15 in.) inside diameter (ID)) and the initial boron concentrations.

The applicant produced test data for four mass flow rate signals in all four loops in the positive flow range without specifying the measurement locations. Additionally, boron concentration data from four signals were reproduced without specifying their measurement locations. A graph with computed loop mass flow rates from the U.S. EPR S-RELAP5 analysis of the 20 cm^2 (3.10 $in.^2$) break case was presented as well.

Based on information obtained at at an audit on March 24 and 25, 2010, and following a discussion at a public meeting on July 15, 2010, in RAI 403, Question 15.06.05-63, the staff requested that the applicant present the available experimental database pertaining to boron dilution relevant to the U.S. EPR boron dilution analysis. The staff considered the database, if limited only to Primärkreislauf-Versuchsanlage Test F1.1, to be insufficient. The staff requested that the applicant clarify why experimental observations and, in particular, PKL findings can be applied to justify the assumptions and conditions used in the analysis of the U.S. EPR under critical LOCA conditions of interest. Also, the staff requested that the applicant clarify how the entire matrix of relevant tests conducted at PKL or other facilities validates the assumptions in the boron dilution analysis including: (1) Restart of natural circulation in only one loop at a time; (2) initial restart in a loop without safety injection; and (3) the boundary conditions assumed in the computational fluid dynamics (CFD) mixing analysis of core inlet boron concentration, including, in particular, the slug injection rate deduced from the PKL experiments. **RAI 403, Question 15.06.05-63, is being tracked as an open item.**

In RAI 167, Question 15.06.05-37, the staff requested that the applicant clarify the extent of mixing the unborated slug and the ECCS injection in the cold leg. In addition, the staff requested that the applicant clarify the relevancy of the CFD calculations, which were performed for Olkiluoto-3, to the U.S. EPR was found necessary.

In a May 28, 2009, response to RAI 167, Question 15.06.05-37, the applicant presented results from selected fluid mixing analyses previously performed for the Olkiluoto 3 (OL3) nuclear power plant with the STAR-CD CFD code. The purpose of the analyses was to examine the mixing process between the de-borated water of moving slugs, the “ambient” borated water present in the cold legs, downcomer, and lower plenum, and the highly borated water from the SIS and the EBS.

The computational domain modeled in the analyses was limited to all four cold legs, the reactor vessel downcomer, and the vessel lower plenum up to the core inlet. The applicant stated that the boundary conditions for the OL3 CFD analyses were based on the PKL test and system analysis results. The PKL test results discussed in the April 30, 2009, response to RAI 167, Question 15.06.05-36, were used to define the volume of the slug, the slug flow rate profile at the restart of natural circulation, and the initial boron concentrations in the slug. The system analysis was performed for OL3 from the beginning of a LOCA until the restart of natural circulation to determine the points in time when restart occurred and associated RCS conditions, such as temperature and pressure.

The cases that were selected as representative for the U.S. EPR and discussed in the June 1, 2009, response to RAI 167, Question 15.06.05-37, involved the same cold leg flow boundary conditions. In the broken cold leg, Cold Leg 3, neither inflow nor outflow were modeled. EBS injection was assumed present in the remaining three intact cold legs, but MHSI flow was modeled in two of the intact cold legs only, Cold Leg 2 and Cold Leg 4.

The runs presented in the June 1, 2009, response to RAI 167, Question 15.06.05-37, pertained to two different cases, identified as Case 3 and Case 4, which differed only with respect to modeling assumptions regarding mixing of MHSI and EBS flows in the cold legs. In Case 3, complete mixing of the MHSI and EBS fluid with the slug at the entrance to the downcomer was assumed. In contrast, a stratified flow in the cold leg was assumed in Case 4 by including separated inlets to the downcomer for the slug and for the MHSI and EBS flow in the model. For each case, three sets of runs, Set 1 through Set 3, with different slug volume and boron concentration assumptions were analyzed.

Set 1 runs assumed a single de-borated slug with a fixed volume of 10.97 m^3 (388 ft^3) and a fixed boron concentration of 50 ppm. Set 2 and Set 3 runs modeled two diluted slugs of different volumes and boron concentrations. The volume of the first slug, which was assumed to accumulate in the SG downside U-tubes, SG exit plenum and crossover pipe, was 18.10 m^3 (639 ft^3), while the volume of the second slug, which was assumed to form in the upward side of the SG U-tubes and SG inlet plenum, was 13.59 m^3 (480 ft^3). The boron concentration of the first slug was set at 50 ppm for Set 2 and at 400 ppm for Set 3, while the second slug was assumed at 685 ppm in both Set 2 and Set 3.

The selected results from the STAR-CD analyses for OL3, as presented in Table 15.6.5-37-2 of the May 28, 2009, response to RAI 167, Question 15.06.05-37, revealed that regardless of the opposing assumptions for mixing of the slug fluid with the SI and EBS flow in the cold leg, Case 3 and Case 4 runs had relatively small differences in the predicted core inlet boron concentrations with the two slugs modeled (14 ppm difference for Set 2 runs and 27 ppm difference for Set 3 runs). With a single slug modeled in Set 1 runs, the difference between Case 3 and Case 4 results was somewhat bigger and amounted to 411 ppm. The predicted minimum boron concentration at the core inlet is 1,987 ppm boron for the dual-slug configuration with a separated cold leg flow model (no MHSI or EBS fluid mixing) and minimum boron content in slug water. Given the staff's experience with and understanding of the modeled flow and geometric configuration, the staff finds the applicant's results reasonable.

Regarding the validity of the presented OL3 results for the U.S. EPR design, the applicant stated that there were no differences between the OL3 geometry used in the STAR-CD simulations and the U.S. EPR RCS. The staff compared the geometry of the OL3 reactor, as modeled in the STAR-CD simulations, to that of the U.S. EPR, and agrees that there are no differences. Additionally, the applicant noted that the boron concentrations in the IRWST, the accumulators, and the EBS tanks in the U.S. EPR reactor, expressed in equivalent units of natural boron, are higher than the values used in the OL3 STAR-CD CFD analyses. The corresponding values were compared in Table 15.6.5-37-3 of the May 28, 2009, response to RAI 167, Question 15.06.05-37, and the staff agreed that this was a valid conservative aspect in the analysis results presented.

The applicant stated correctly that the diluted slug volumes assumed in the analyses were significantly larger than the volume of the crossover pipe in the U.S. EPR design and referred to the predictions from the S-RELAP5 analyses that the crossover pipe did not completely fill prior to the refill stage. Based on this condition, the applicant concluded that even with larger slug volumes and system boron concentrations lower than those used in the U.S. EPR, the STAR-CD CFD results predicted a degree of mixing in the reactor vessel sufficient to raise the minimum boron concentration at the core inlet above the core-wide threshold value necessary for maintaining a one-percent core shutdown margin.

The staff examined CFD calculations related to the boron dilution issue during a January 26, 2010, audit. The 1.2 million node model represents the four cold legs, downcomer, and lower plenum up to the elevation of the core. The STAR–CFD code was used. The geometry represents the Olkiluoto 3 reactor. Boundary conditions are taken from PKL and CATHARE. The calculations show that for an 11 m^3 (388 ft^3) slug at 50 ppm initial boron concentration, the boron concentration at the core inlet will be greater than the minimum value needed to prevent core criticality. Visual representation of the calculations shows the extensive mixing in the downcomer. The calculations also showed that neglecting mixing in the cold leg does not significantly change the boron concentration at the core inlet. Accordingly, the staff considers RAI 167, Question 15.06.05-37, resolved.

In RAI 167, Question 15.06.05-42, the staff requested that the applicant clarify the sensitivity of the boron dilution analysis to the initial core stored energy. In a March 31, 2009, response to RAI 167, Question 15.06.05-42, the applicant addressed the effect of the initial core stored energy under-prediction with respect to boron dilution. The applicant stated that, in the SBLOCA boron dilution analysis, the accumulation of condensate is predicted to occur at approximately 500 sec after the break initiation for the larger break sizes of approximately 0.1016 m ID (4.0 in. ID) and 3,000 sec for the smaller break sizes of approximately 0.0381 m ID (1.5 in. ID). According to the applicant, however, the dissipation of the fuel initial stored energy was predicted to occur between 50 sec for the larger break sizes and 150 sec for the smallest break sizes. Thus, the applicant concluded that the only source of heat contributing to the generation of steam and the subsequent condensate accumulation downstream of the SG was the decay heat and not the initial core stored energy. The applicant stated that the stored energy had an insignificant impact on the amount of condensate accumulation in the crossover pipes and consequently on the boron dilution effect and the staff concurred with this conclusion because the times of interest for boron dilution are far longer than the times to dissipate stored energy. Accordingly, the staff finds the applicant’s March 31, 2009, response acceptable. Therefore, the staff considers RAI 167, Question 15.06.05-42, resolved.

Following an LBLOCA, there must be a sufficient mass of borated water available from the IRWST to ensure long-term cooling and that the core remains subcritical. Furthermore, alternate sources of boron free water should not be injected into the RCS from external sources. During an audit at AREVA headquarters in Lynchburg, VA on March 24, 2010, and March 25, 2010, the staff examined the spreadsheet calculations contained in the applicant’s detailed analysis. The analysis assumes no boron in the RCS and that the injection sources, namely the accumulators and the IRWST, are at the minimum concentration of 1,700 ppm as would be required by TS 3.5.1 and 3.5.4. In order to inject unborated water, the MHSI or LHSI pumps would need to be aligned to an unborated water source, rather than the IRWST. In addition, the staff reviewed FSAR Tier 2, Figures 6.3-1, “Safety Injection System Overview,” 6.3-2, “Safety Injection/Residual Heat Removal,” and 6.3-3, “IRWST Layout.” These piping and instrumentation diagrams (P&ID) show no means to realign the SI pumps to an unborated water source. A source of unborated water is available from the CVCS, however. Automatic interlocks are provided to prevent inadvertent operation of the CVCS; therefore, the staff concludes that injection of unborated water is precluded.

15.6.5.3.4.3 *Long-Term Cooling and Boron Precipitation Prevention*

Following a LOCA, during the pool boiling phase that follows the initial mitigation of all but the smallest LOCAs, as discussed in Section 15.6.5.3.2, boron could concentrate in the reactor core region if countermeasures are not taken. If the concentration of boron reaches the solubility limit, boron precipitates out of solution and may cause coolant channel blockage which could result in degraded heat removal from the fuel. The U.S. EPR incorporates two design features intended to prevent the concentration of boron from reaching the solubility limit:

- MSRTs that depressurize the SGs following an SBLOCA, thus cooling and depressurizing the RCS primary side to increase SI flow and re-establish natural circulation. A partial cooldown is initiated automatically in response to an SI signal and depressurizes the SGs down to 870 psia at a rate of 82.2 °C/hr (180 °F/hr). An operator action is credited to continue the initial partial cooldown of the SGs at a lower rate of 32.2 °C/hr (90 °F/hr). This allows the MHSI and LHSI systems to refill the RCS and restart natural circulation if the break is small enough.

- Provisions for redirecting part of the LHSI flow to the RCS hot leg. The U.S. EPR design employs hot-leg injection to control boric acid concentration in the core during pool boiling following a LOCA. FSAR Tier 2, Section 6.3 states that the availability of four separate hot-leg connections, one for each of the SIS trains, preserves the hot-leg injection function for such purposes. The LHSI system can be manually re-aligned during the accident recovery phase for hot-leg injection to both prevent boron precipitation and mitigate steaming from the break. The simultaneous injection into both hot and cold legs flushes the core and prevents the concentration of boric acid from reaching solubility limits.

FSAR Tier 2, Section 15.6.5.4.1, “Prevention of Boric Acid Precipitation,” presents the boron precipitation analysis results for both small-break and large-break LOCAs. It explains that the U.S. EPR provides the operator the capability to redirect an LHSI train so that at least 75 percent of its flow is injected through the hot leg letdown line of the residual heat removal system (RHRS) with the exact fraction of the hot leg flow being dependant on the RCS pressure. This FSAR section also states that the LBLOCA bounds the SBLOCAs with regard to boron precipitation inasmuch as the analyses show that more water is retained in the core region in the case of SBLOCAs. This can be also seen in FSAR Tier 2, Figure 15.6-92, “Time Dependent Boron Concentration During the Pool Boiling Period,” which shows the calculated time dependent boron concentrations during the long-term cooling pool boiling period. The mitigating effect of hot leg injection is confirmed by extending the S-RELAP5 calculations for a representative range of breaks analyzed in FSAR Tier 2, Sections 15.6.5.1 and 15.6.5.2. FSAR Tier 2, Sections 15.6.5.4.1.1 and 15.6.5.4.1.2 discuss the U.S. EPR thermal-hydraulic response and flow behavior important for boron precipitation for SBLOCAs and LBLOCA, respectively.

FSAR Tier 2, Section 15.6.5.4.1.1, “Small-Break LOCA Flow Behavior,” summarizes the technical basis of the boron precipitation analysis and presents the results of that analysis for the limiting SBLOCA case. The following summarizes the applicant’s analysis. The applicant examined five SBLOCA cases with break sizes ranging between 3.81 cm (1.5 in.) and 16.51 cm (6.5 in.) in equivalent diameter, and determined that for breaks up to 10.16 cm (4 in.) in diameter, the RCS is refilled in less than 4 hours and returns to natural circulation with two trains of MHSI and LHSI in operation. For such breaks, boric acid buildup and the associated possibility for boron precipitation is precluded upon resumption of natural circulation. In the analysis of the limiting case, which is a 16.51 cm (6.5 in.) break, two operator actions were credited at 1,800 sec following completion of the initial automatic partial cooldown. These are: (1) The operator takes action to continue depressurization of the SGs at a rate of 50 °C/hr (90 °F/hr) and (2) the operator realigns both operating LHSI trains to inject approximately 75 percent of their flow into the corresponding hot legs. The analysis results in FSAR Tier 2, Figure 15.6.5-84, depict the integrated mass flows from the upper plenum to all four hot legs. FSAR Tier 2, Figure 15.6.5-85 shows the integrated core region exit mass flows for the inner, average, and peripheral core regions along with the hot assembly. FSAR Tier 2, Figure 15.6.5-86, plots the integrated mass flow from the reactor vessel lower head to the lower plenum. The results show that the redirected LHSI flow reverses the hot leg flow direction in the corresponding loops into the upper plenum, thus delivering additional coolant to the core region as seen from FSAR Tier 2, Figure 15.6.5-84. In addition, the LHSI flow into the hot legs further reverses the flow through the fuel assemblies in the peripheral core region as shown in FSAR Tier 2, Figure 15.6.5-85, with some of the downward flow penetrating through the lower plenum into the lower head region, as shown in FSAR Tier 2, Figure 15.6.5-86. Based on these predictions, in FSAR Tier 2, Section 15.6.5.4.1.1, the applicant concluded that the LHSI switchover to hot leg injection prevents boron precipitation in the core and the RCS by inducing the described major recirculation flow pattern.

FSAR Tier 2, Section 15.6.5.4.1.2, “Large-Break LOCA Flow Behavior,” discusses the effectiveness of switchover to hot leg injection for breaks that are too large in size to allow for the MHSI and LHSI to refill the loops. In the analysis of the representative cold leg LBLOCA case, the applicant assumed that the operator took action to switch to hot leg injection at 2,000 sec. Although the U.S. EPR design can redirect 75 percent of the LHSI through the hot leg, the applicant assumed that the switch redirected only 50 percent of the LHSI flow. The applicant presented analysis results in FSAR Tier 2, Figure 15.6.5-88, which depicts the integrated mass flows from the upper plenum to a hot leg of a loop with LHSI flow. Further, FSAR Tier 2, Figure 15.6.5-89 shows the integrated core region exit mass flow for the peripheral core region. FSAR Tier 2, Figure 15.6.5-90 plots the integrated mass flow from the lower plenum onto the peripheral core region, and FSAR Tier 2, Figure 15.6.5-91 illustrates the integrated mass flow from vessel lower head to the lower plenum. As in the SBLOCA case discussed above, the results in FSAR Tier 2, Figure 16.6.5-88 show that LHSI water injected into the two hot legs receiving LHSI flows into the reactor upper plenum. Furthermore, FSAR Tier 2, Figure 15.6.5-89 indicates a downward flow reversal through the peripheral core region at about 2,200 sec, while the flow remains positive in the inner core regions. FSAR Tier 2, Figure 15.6.5-90 also demonstrates a flow reversal from the peripheral fuel assemblies into the lower plenum and FSAR Tier 2, Figure 15.6.5-91 shows that the flow from the lower head into the lower plenum reverses at about 2,250 sec, which indicates an overall flow reversal in the reactor vessel. The applicant stated in the FSAR that this recirculation flow would flush excess boron from the core and the vessel to the cold-leg break. In the case of a hot leg LBLOCA, FSAR Tier 2, Section 15.6.5.4.1.2 states that ECCS injection into the cold leg exceeds the core boil off rate. Thus, the applicant concludes that the fraction of the ECCS injection in excess of the core boil off rate would produce sufficient flow through the core to prevent the increase of boron concentration to levels that approach the precipitation limit even with redirection of half of the LHSI flow to the hot legs.

The staff notes that the results for a representative LBLOCA case were presented in the FSAR to demonstrate the effectiveness of hot leg injection for break sizes too large for the MHSI and LHSI to refill the loops. The results predicted that flow reversal would occur at about 2,200 sec in the peripheral fuel assemblies of the core with the flow remaining positive in the central core regions due to higher power density and heating. This flow regime suggests effective mixing of the coolant in the core. In addition, flow reversal from the RPV lower plenum into the lower head was predicted at about 2,250 sec, which indicates overall flow reversal in the RPV. Since boron will not precipitate until the concentration reaches about 38,000 ppm, and the initial concentrations are about 2,000 ppm, the staff concludes that this flow reversal will flush excess boron from the core and vessel to the cold-leg break.

FSAR Tier 2, Section 15.6.5.4.1.3, "Boron Precipitation Assessment," states that in both SBLOCA and LBLOCA analyses, the maximum initial boron concentration in the concentrating volume was determined by mixing the conservatively biased solutions in the IRWST, accumulators, and the manually actuated EBS. The maximum boron concentration derived from this procedure was 1,929 ppm. This FSAR section cites AREVA Technical Report ANP-10288P, "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report," Revision 1, January 2010, for a more detailed description of the U.S. EPR boron precipitation analysis methodology. This FSAR section also refers to FSAR Tier 2, Figure 15.6-92, "Time Dependent Boron Concentration During the Pool Boiling Period," which depicts the predicted boron concentration histories for the limiting LBLOCA PCT case and the bounding 16.51 cm (6.5 in.) diameter SBLOCA. Furthermore, the applicant states that the results for both representative cases demonstrate boric acid precipitation is prevented for the complete spectrum of breaks with assumed timing of switching to hot leg injection at 2,000 sec for the LBLOCA case and at 1,800 sec for the SBLOCA case. The figure also shows that the solubility limit is reached at about 6,800 sec for the LBLOCA case and at about 16,360 sec for the SBLOCA case if no operator action is taken to initiate hot-leg injection. Thus, the applicant concludes that there is adequate time for the operator to initiate hot leg injection to limit the buildup of boron in the core region and prevent precipitation in other regions of the RCS.
To confirm these conclusions, the staff reviewed the supporting boron precipitation analysis methodology and results provided in ANP-10288P as discussed below.

The U.S. EPR quantitative analysis of boron precipitation was described in detail initially in AREVA Technical Report ANP-10288P, "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution," November 2007, Revision 0. Therefore, the staff reviewed pertinent portions of this report. The boron precipitation analyses presented in the report used input from S-RELAP5 analyses of both small-break and large-break LOCAs performed to evaluate the plant thermal-hydraulic response during the post-LOCA period. Cases that are potentially susceptible to boron precipitation were analyzed to provide input for bounding boron precipitation calculations to ensure that boron concentrations remain within solubility limits. The results from the U.S. EPR boron precipitation analysis methodology presented in ANP-10288P, Revision 0, were performed with a single-region concentrating control volume assuming perfect mixing and using a spreadsheet.

The SBLOCA cases were evaluated in ANP-10288P, Revision 0, using the approved S-RELAP5-based methodology described in AREVA NP Report, "Codes and Methods

Applicability Report for the U.S. EPR," ANP-10263P-A, August 2007. For break sizes up to 10.16 cm (4 in.) diameter, two trains of ECCS can refill the RCS and establish single-phase natural circulation within 4 hours. This capability limits the period of pool boiling to less than 3 hours. A 5.08 cm (2 in.) diameter break size case was analyzed as representative of this range of break sizes. Considering small breaks for which the ECCS is predicted to establish two-phase natural circulation, it was stated in ANP-10288P, Revision 0, that a 15.24 cm (6 in.) diameter break is small enough to allow two trains of the ECCS to partially refill the active loops of the RCS after depressurization, but too large for the establishment of single phase natural circulation. For such a break, operator action is credited at 1,800 sec to initiate cooldown at 32.2 °C/hr (90 °F/hr). The collapsed liquid level is predicted to begin to increase in the hot-leg side of the SGs of the two loops receiving hot-leg injection. The applicant predicts that approximately 2,000 sec later, the collapsed liquid level also begins to increase in the cold-leg sides of the tube bundles. The two active loops refill sufficiently at 12,500 sec to establish two-phase natural circulation, while the two inactive loops do not. The applicant indicated that the two-phase natural circulation would mix the coolant in the active loops to distribute the boron and prevent re-concentration of boron in the core. In ANP-10288P, Revision 0, the applicant states that breaks larger than about 15.24 cm (6 in.) in diameter are too large for two trains of LHSI to refill the loops and establish two-phase natural circulation. Sequences involving such breaks are predicted to remain in pool boiling.

The applicant states that the post-LOCA phenomena associated with boron concentration in the core is independent of break size once the break is too large for the LHSI to refill the loops. The LBLOCA cases were evaluated in ANP-10288P, Revision 0, using the realistic large-break LOCA methodology based on S-RELAP5 and described in AREVA Topical Report, "U.S. EPR Realistic Large Break Loss of Coolant Accident Topical Report," ANP-10278P, Revision 0, March 2007. ANP-10288P, Revision 0, referred to an 0.1858 m^2 (2.0 ft^2) LBLOCA case as being representative of this class of breaks. In this case, operator actions at 2,000 sec to continue the partial cooldown of the SGs at 32.2 °C/hr (90 °F/hr) and redirect 50 percent of the LHSI system injection to the hot-leg RHRS nozzles were credited. The analysis results presented in ANP-10288P, Revision 0, included plots of the integrated mass flow from the upper plenum to one of the hot legs receiving LHSI water, integrated mass flow from the average core region to the upper plenum, and integrated mass flow from reactor vessel lower head to the lower plenum. The results demonstrated that the hot-leg injection purged the core region by removing concentrated boric acid solution that accumulated before the initiation of hot leg injection. For a large break in a hot leg, ANP-10288P, Revision 0, states that the 50 percent of LHSI flow that is injected into the cold legs is sufficient to flush the core coolant toward the break. As stated above, FSAR Tier 2, Section 15.6.5.4.1 describes that the U.S. EPR provides the operator the capability to redirect an LHSI train so that at least 75 percent of its flow is injected through the hot leg RHRS letdown line. Further, ANP-10288P, Revision 1, clarified that for a large break in a hot leg, the MHSI and remaining portion of LHSI that is injected into the cold legs after the switch to hot leg injection is sufficient to flush the core coolant toward the break. In addition, the applicant stated that the LHSI injected into the intact hot legs also is available to the core to prevent excessive boron concentration. The staff concludes that the remaining available injection flow is much greater than the boil-off rate and, therefore, the staff concurs in the applicant's analysis.

Calculations in ANP-10288P, Revision 0, showed initially that to avoid boron precipitation, there is about 2 hours to initiate hot leg injection for the limiting LBLOCA. The earlier switchover time provides considerable margin for boron precipitation in the AREVA calculations. In order to better understand the quantitative analysis presented in ANP-10288P, Revision 0, the staff issued several RAIs, the details of which are described below.

In RAI 30, Question 15.06.05-1, the staff requested that the applicant provide detailed information about the U.S. EPR geometry, loop friction, pressure losses, mass flow rates, flow areas, k-factors, hydraulic diameters, and coolant temperatures. The staff also requested that the applicant provide the mixing volume void fraction, LHSI head flow curve, and certain core and piping elevations. The intent of the request was to obtain sufficient detailed information to permit independent staff calculations of long term cooling. The applicant provided the requested information in the September 24, 2008, response to RAI 30, Question 15.06.05-1. In a September 24, 2008, response to RAI 30, Question 15.06.05-1, the applicant provided information needed to enable the staff to perform independent calculations or verify results obtained by the applicant as necessary for the review process. Accordingly, the staff considers RAI 30, Question 15.06.05-1, to be resolved. The staff calculations are discussed below.

In RAI 30, Question 15.06.05-2, the staff requested that the applicant explain if the LBLOCA precipitation times and associated timing to switchover to hot leg injection includes breaks located on the top of the cold leg piping. In an August 28, 2008, response to RAI 30, Question 15.06.05-2, the applicant stated that the LOCA analysis is based on a side pipe opening. The core region void fraction is determined by summing the volume of the void in each node of each region of the S-RELAP5 analysis in the core mixing/concentrating region at 200 sec for the 0.1905 m^2 (2.05 ft^2) RLBLOCA. The applicant stated that this is used as a conservative bound for the analysis of LBLOCA boron precipitation, which sets the maximum time allowed before simultaneous injection is initiated. In addition, the applicant stated that no credit was taken for the increase in water content in the core concentrating region as the decay heat decreases with time. The staff finds that neglecting the effect of decreasing decay heat on the liquid content in the mixing volume is conservative from the point of view that reduced decay heat generation will produce less voiding in the core region, thus increasing the liquid inventory in the core region. While finding the analysis approach reasonable, the staff determined that the applicant response did not address how the results and the conservatism of the analysis will be affected by assuming a break location at the top of the discharging leg piping. The staff addressed this issue as part of RAI 403, Question 15.06.05-64, as discussed below.

In RAI 30, Question 15.06.05-3, the staff requested that the applicant clarify the sump temperature versus time following initiation of recirculation and how that effected boron precipitation. In an August 28, 2008, response to RAI 30, Question 15.06.05-3, the applicant explained that the analyses assumed that the solution enters the core region at the saturation temperature at atmospheric pressure (i.e., 100 °C (212 °F)) with all of the decay heat going into the latent heat of evaporation. As stated in the response, accounting for the sump temperature and cooling by the low head safety injection heat exchangers would decrease the steam generation rate and steam flow exiting the core concentrating region. Since neglecting core inlet flow subcooling increases the core steaming rate, assuming saturated temperature for the core inlet water in the analyses represented in Figure 2-19 of ANP-10288P, Revision 0, is conservative and the staff considers it acceptable.

In RAI 30, Question 15.06.05-4, the staff requested that the applicant clarify if debris from the sump could block portions of the core inlet, and if so, what the impact on precipitation timing in the regions where the core boric acid cannot diffuse downward into the lower plenum will be. In an August 28, 2008, response to RAI 30, Question 15.06.05-4, the applicant stated that, with the limited fibrous material available in the U.S. EPR, there is no downstream blockage that can impact core cooling. The extent and impact on core cooling by debris entering the core is considered in AREVA Technical Report ANP-10293P. To account for the results from separate effects testing of a simulated U.S. EPR fuel assembly in addressing in-vessel downstream effects, the applicant confirmed in an April 28, 2011, letter to the NRC (NRC:11:037), that an updated, conforming response to RAI 30, Question 15.06.05-4, will be prepared and submitted. To follow up on this update, in RAI 493, Question 15.06.05-99, the staff restated the questions asked in RAI 30, Question 15.06.05-4, requesting that the applicant consider if debris from the sump could block portions of the core inlet and if so, evaluate the impact on precipitation timing in the regions where the core boric acid cannot diffuse downward into the lower plenum. The staff also requested that the applicant identify the maximum core inlet blockage that can occur, show local concentrations in the core are below the precipitation limit, and show that with the core inlet blocked, and boric acid and other precipitates in the core, that the switch to simultaneous injection can flush the core and reduce the concentration to acceptable levels. **RAI 493, Question 15.06.05-99, is being tracked as an open item.**

In RAI 30, Question 15.06.05-5, the staff requested that the applicant clarify what happens to the boric acid in the vapor as it passes through the SGs to reach the break. In a September 24, 2008, response to RAI 30, Question 15.06.05-5, the applicant indicated that boric acid plate-out in the SG was estimated based on the stored heat capacity of the secondary water and tube metal. The applicant described the phenomenology as follows: The secondary water will stratify, as the cooling is from the bottom where the droplets and steam enter the tube bundle. The applicant stated that the calculated mass of droplet water evaporated multiplied by the specific heat of vaporization equals the energy removed to cool the unit tube cell from 282.2 °C (540 °F) to 100 °C (212 °F), assuming the steam is not superheated as it passes through the SG; that is, the SG cell energy goes into droplet evaporation on the tube surfaces. The applicant stated that the boric acid in the water droplets that evaporate to cool the unit cell would plug about 12 percent of the tube flow area at any location in the SG if the droplets were at the boron precipitation limit of 38,500 ppm at 100 °C (212 °F). The staff finds that this method conservatively estimates the SG blockage because it maximizes precipitation, and is therefore acceptable. ANP-10288P, Revision 0, also states that the U.S. EPR boron concentration analysis does not credit removal of boric acid in steam transported out of the core, and nor does it credit carryout of boron from the reactor vessel due to droplet entrainment. The staff agrees that those are robust conservative assumptions as they maximize the boron content in the boron concentrating volume. Accordingly, the staff considers RAI 30, Question 15.06.05-5, resolved.

In RAI 30, Question 15.06.05-6, the staff requested that the applicant justify and clarify how the void fractions were determined in ANP-10288P, Table 2-3 "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report," Revision 0, November 2007. In an August 28, 2008, response to RAI 30, Question 15.06.05-6, the applicant stated that the void fractions in ANP-10288P, Revision 0, Table 2-3 are determined by dividing the sum of the node void volumes by the sum of the total node volumes from each component/region in the S-RELAP5 analysis in the core mixing/concentrating region at 200 sec into the transient for the 0.1858 m^2 (2 ft^2) RLBLOCA. The applicant explained that there were two typographical errors in Table 2-3 (in the void fraction column) in ANP-10288P, Revision 0, and provided the corrected values in Table 15.06.05-6-1 in the response. The applicant also clarified that the region total volume and the liquid volumes in the last column of Table 2-3 in ANP-10288P, Revision 0, were correct. The staff finds the explanation and the corrections provided in the response acceptable. Additional information regarding the basis for choosing the 200 sec time for void fraction is provided in the response to RAI 30, Question 15.06.05-12, considered below.

In RAI 30, Question 15.06.05-7, the staff requested that the applicant clarify if the mixing volume considered the maximum content of sump debris that can accumulate in the core. In an August 28, 2008, response to RAI 30, Question 15.06.05-7, the applicant indicated that the mixing volume does not consider the maximum content of sump debris that can accumulate in the core. Further, the applicant stated that the maximum amount (volume) of debris that can accumulate in the core and lower plenum regions during recirculation will be based on future testing. As in the case of RAI 30, Question 15.06.05-4, addressed previously, the extent and impact on core cooling by debris entering the core is considered in ANP-10293P. Therefore, in follow-up RAI 493, Question, 15.06.05-100, the staff requested that the applicant consider the maximum content of sump debris that can accumulate in the core, and identify the maximum amount (volume) of debris that can accumulate in the core and lower plenum regions during recirculation. **RAI 493, Question 15.06.05-100, is being tracked as an open item.**

In RAI 30, Question 15.06.05-8, the staff requested that the applicant provide a description of the tests used to validate the two-phase level swell and boric acid precipitation models. In a September 24, 2008, response to RAI 30, Question 15.06.05-8, the applicant stated that S-RELAP5 was used to calculate level swell and referenced the previously approved code validation described in EMF-2103(P)(A), "Realistic Large Break Loss-of-Coolant Accident Methodology for Pressurized Water Reactors," April 9, 2003. The applicant referred to Full Length Emergency Cooling Heat Transfer (FLECHT) Skewed, Full Length Emergency Cooling Heat Transfer - Separate Effects and System Effects Test (FLECHT-SEASET), Thermal Hydraulic Test Facility (THTF) level swell, General Electric (GE) level swell, FRIGG (a Swedish facility), Slab Core Test Facility (SCTF), Cylindrical Core Test Facility (CCTF), and Upper Plenum Test Facility (UPTF) carry-out test validations. With regard to the U.S. EPR boric acid precipitation model, the applicant clarified that the boron model in S-RELAP5 was not used in the precipitation analysis. Instead, S-RELAP5 was only used to determine the concentrating water volume at 200 seconds and to show that the switch to simultaneous injection for the 0.381 m^2 (4.1 ft^2) LBLOCA resulted in dilution of the boron concentration accumulated before the switch. The boron precipitation concentration was defined to bound the 100 °C (212 °F) mixing limit for water in the core and cold solution in the IRWST at its minimum technical specification limit of 15 °C (59 °F) in accordance with FSAR Tier 2, Table 6.3-4, "IRWST Parameters." The applicant also stated that the boron precipitation model conservatively neglected the following factors: (1) Increase of water inventory with time in the concentrating region; (2) subcooling of the water entering the core concentrating region; (3) boron solubility in steam; (4) droplet entrainment leaving the core concentrating region and removing boron; (5) plate-out on the upper internals surfaces in the core concentrating region due to droplet evaporation; (6) increased solubility of boron due to other solutes; (7) increase in boiling temperature due to boric acid concentration; and (8) water in the upper plenum above the bottom of the outlet nozzles. The decay heat was modeled with draft ANS 5.1 1971 standard using a multiplier of 1.2 for the fission product decay heat in accordance with 10 CFR Part 50, Appendix K and as approved for the existing fleet of operating plants. The staff finds that the identified modeling assumptions implemented in the U.S. EPR boron precipitation analysis are robustly conservative in that each factor neglected in the analysis would result in increased boron solubility, and these modeling assumptions are therefore acceptable. Accordingly, the staff considers RAI 30, Question 15.06.05-8, resolved.

In RAI 30, Question 15.06.05-9, the staff requested that the applicant justify the mixing of boric acid from the core into the lower plenum region throughout the entire time analyzed. In an August 28, 2008, response to RAI 30, Question 15.06.05-9, the applicant stated that the lower plenum sub-volume given in ANP-10288P, Revision 0, Table 2-3 includes the volume from the bottom of the lower core support plate to the beginning of the fueled section in the fuel pin inside of the core barrel. It does not include any of the lower head volume. The difference in the heat generation distribution in the core region is expected to induce turbulence at least this far down into the region inside the core barrel. Once the hot side injection is started, the turbulence and recirculating flow at the lower support plate are expected to make the lower head volume part of the mixing volume. Additional information regarding the mixing volume is provided in responses to RAI 241, Question 15.06.05-54, and RAI 241, Question 15.06.05-55, which are described below. The staff finds it appropriate to include only a small fraction of the lower plenum from the bottom of the lower core support plate to the beginning of the heated core regions inside of the core barrel into the concentrating mixing volume, considering the described mixing processes. Therefore, the staff finds that it is reasonable and acceptable to include this sub-volume as part of the mixing volume. Accordingly, the staff considers RAI 30, Question 15.06.05-9, resolved.

In RAI 30, Question 15.06.05-10, the staff requested that the applicant clarify the appropriateness of combining the low boric acid concentration core coolant with the other water sources in the analysis. The staff requested that the applicant discuss the effect of excluding the initial RCS boric acid concentration from the boric acid precipitation calculations. The staff also noted that use of a mixed mean boric acid concentration to determine precipitation time does not take into account that some of the higher concentration sources could inject at much higher flow rates than assumed in the analysis. The staff requested that the applicant provide the boron concentration versus time, assuming that the EBS injects alone followed by only the IRWST as sources. In a September 24, 2008, response to RAI 30, Question 15.06.05-10, the applicant stated that the EBS flow is insufficient to cool the core and that MHSI and LPSI are automatically actuated. In order to clarify the time-dependent boron injection rate, in RAI 241, Question 15.06.05-33, the staff requested additional information as discussed below.

In RAI 30, Question 15.06.05-11, the staff requested that the applicant clarify the means by which the boric acid from the high concentration EBS tanks can be injected into the RCS and questioned if the EBS could be the sole source of injection once the accumulators empty. In a September 24, 2008, response to RAI 30, Question 15.06.05-11, the applicant explained the operation of the EBS and stated that for LOCAs that empty the accumulators, all four trains of the MHSI and LPSI systems would have to fail for the EBS injection pumps to be the only source of water injection into the RCS. Therefore, it is not postulated that the EBS could be the only solution injection source. The staff finds the explanation provided by the applicant acceptable since failure of all four trains of MHSI and LPSI need not be assumed in accordance with the single failure criterion, and considers RAI 30, Question 15.06.05-11, resolved.

In RAI 30, Question 15.06.05-12, the staff requested that the applicant explain the basis for choosing the 200 sec time for void fraction to compute the boric acid concentration versus time, and the basis for choosing a 0.232 m^2 (2.5 ft^2) break. In an August 28, 2008, response to RAI 30, Question 15.06.05-12, the applicant stated that the 200 second time for void fraction is chosen because it is the end of the run for the specific RLBLOCA PCT case analyzed in ANP-10288P, Revision 0. The boron precipitation analyses are related to long-term cooling, thus using the boron concentration value at 200 sec and holding it constant is conservative for establishing the time to switch to simultaneous injection. The applicant also explained as follows: SBLOCA is evaluated for breaks up to diameters of 15.24 cm (6 in.). All SBLOCA cases return to natural circulation. The 0.372 m^2 (4 ft^2) break is considered to be a reasonable extension for the LBLOCA analyses to establish a time to simultaneous injection, because the water volume at 200 sec is held constant with time through the evaluation. This specific RLBLOCA PCT case chosen is extended to about 6,000 sec and confirms that the value at

200 sec is conservative. In addition, the case extended to 6,000 sec confirms that the simultaneous injection dilutes the solution in the core concentrating region. In the August 28, 2008, response to RAI 30, Question 15.06.05-12, the applicant also explained that the LBLOCA break size of 0.232 m^2 (2.5 ft^2), as found in ANP-10288P, was a typographical error, and the break used in this analysis was 0.190 m^2 (2.05 ft^2) at each side for a 0.381 m^2 (4.1 ft^2) total area. The staff finds the explanation in the response to this question acceptable insofar as the S-RELAP5 results for the considered RLBLOCA PCT case extended to about 6,000 sec confirmed that the 200 second-based void fractions yielded a conservative value for the mixing volume liquid content. However, the staff requested additional information to further address the conservatism in determining the control volume liquid content considering the importance of the parameter. In follow-up RAI 241, Question 15.06.05-52, the staff requested that the applicant clarify the conservatism in determining the control volume liquid content considering the importance of the parameter as discussed below.

In RAI 30, Question 15.06.05-13, the staff requested that the applicant clarify if the switchover to hot leg injection was identified in the emergency operating procedures (EOPs), and the staff also requested that the applicant clarify if the actions could be performed within the time assumed in the safety analysis. In addition, the staff requested that the applicant identify the entrainment rate in the hot legs due to steaming from the core at the lowest achievable RCS pressure following the limiting LBLOCA. The staff requested that the applicant justify and demonstrate that the hot leg injection is not entrained into the hot legs and SGs at 7,200 sec, and justify the maximum elevation of the two-phase level in the hot leg at this time so that the appropriate vapor velocity (used in the entrainment calculation) in the upper portion of the hot leg could be determined. In a September 24, 2008, response to RAI 30, Question 15.06.05-13, the applicant stated that the EOPs, when written, would include direction regarding switchover to hot leg injection. The applicant also explained that an analysis of an LBLOCA with MHSI and LHSI trains injecting into Loops 2 and 3 shows a slight drop in the collapsed liquid level in the core with the switch of about half of the LHSI flow to the hot leg as simultaneous injection. The applicant further explained that the analysis also shows that the penetration of the hot leg injection into the upper plenum and down through the reactor core will prevent accumulation of boron in the core region within about 200 sec after simultaneous injection is initiated. An RLBLOCA case was extended to about 6,000 sec, with simultaneous injection starting at 2,000 sec. Since decay heat decreases with time, an early start increases the likelihood that the injection water will be carried into the SGs. With half of the injected LHSI water going into the hot leg, the water begins to drain back into the upper plenum within 100 sec. The water entering the upper plenum is mixed in the two-dimensional upper plenum model and is carried down through the outer core low power region. Some of the water that is heated with the mixing and down-flow through the core goes into the lower plenum and then into the lower head and into the downcomer. In addition, some energy is carried out with the backflow of water from the core concentrating region, which reduces the quantity of steam generated in the core region. This reduces the quantity of steam to be vented around the loops. The void fraction in the RCS pipe entry node of the hot legs that is receiving simultaneous injection goes from about 60 percent to about 80 percent. As the steam flow decreases with time, the void fraction starts to decrease in that node at about 5,000 sec. The void fraction in the RCS pipe entry node of the loops that are not receiving LHSI injection in the hot leg increases much less, holding at about 60 percent, and starts to decrease more rapidly around 5,000 sec. These analyses show that the entrainment does not negate hot leg injection for times greater than 2,000 sec. The staff accepted the explanation provided in this response with regard to addressing the time to switch to simultaneous injection in the EOPs as the response stated that this will be done in a manner consistent with safety analysis case and the switch will take less than 10 minutes as it can be made from the control room. Nonetheless, with regard to the remaining aspects of the question, the staff determined that the applicant response did not address sufficiently the concern related to possible ECC flow entrainment in the hot leg. The staff addressed this issue as part of RAI 403 as discussed below.

Following an onsite reactor systems audit held in Lynchburg on April 21-24, 2009, the staff identified further thermal-hydraulic areas that called for additional information to allow the staff to complete the review of the SBLOCA and LBLOCA boron precipitation analyses documented in ANP-10288P, "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report," Revision 0, November 2007. Accordingly, the staff issued RAI 241, Questions 15.06.05-52 through 15.06.05-55, which related to boron precipitation. In a November 25, 2009, response to RAI 241, Questions 15.06.05-52 through 15.06.05-55, the applicant provided the responses as discussed below.

In RAI 241, Question 15.06.05-52, the staff requested that the applicant clarify the appropriateness of the void fraction values applied for each of the sub-volumes within the control mixing volume, which included the heated core region, heavy reflector, guide tube regions, lower support plate to heated core (lower plenum), and upper plenum up to the bottom of the hot legs. In the SBLOCA analysis, the void fractions were based on S-RELAP5 predictions for a 15.24 cm (6 in.) diameter SBLOCA case at 20,000 sec and were given in ANP-10288P, Table 2-2, "U.S. EPR Post-LOCA Boron Precipitation and Boron Dilution Technical Report," Revision 0, November 2007. In the LBLOCA analysis, the void fractions were set equal to the values predicted at the end of the 0.190 m^2 (2.05 ft^2) realistic LBLOCA case with maximum PCT. The case modeled a 0.190 m^2 (2.05 ft^2) opening area on each side of the break for a total break area of 0.381 m^2 (4.1 ft^2)), as discussed above with reference to RAI 30 Question 15.06.05-12. These values were provided in Table 2-3 of ANP-10288P and corrected in Table 15.06.05-6-1 in the August 28, 2008, response to RAI 30, Question 15.06.05-6.

In a November 25, 2009, response to RAI 241, Question 15.06.05-52, the applicant stated that all five sub-volumes were split into more nodes in the S-RELAP5 LBLOCA model, and the liquid contents in the sub-volumes were calculated using all of the nodal void fractions in each particular sub-volume, which resulted in a total volume of 17.47 m^3 (617 ft^3) as given in Table 2-3 in ANP-10288P, Revision 0. Furthermore, the applicant asserted that the LBLOCA is more limiting than the SBLOCA for the purposes of boron precipitation analysis due to the lower liquid volumes in the core regions, and based its November 25, 2009, response to RAI 241, Question 15.06.05-52, on U.S. EPR S-RELAP5 realistic LBLOCA analysis results. The applicant described this analysis as follows: In the realistic S-RELAP5 LBLOCA model, the core region is modeled by 27 axial nodes and four radial regions. The volumes at the end of the PCT transient for all 118 cases (59 cases for cycle 1 and 59 cases for the equilibrium cycle) of the RLBLOCA analysis were examined to identify the runs that contained the least water volume in the concentrating region. These three LBLOCA cases were rerun to extend the transient analyses to 6,000 sec, and the results for the predicted water inventory in the concentrating region were documented in Figure 15.06.05-52-1 in the November 25, 2009, response to RAI 241, Question 15.06.05-52. While the concentrating volume liquid inventory predicted at the end of the PCT analysis (200 sec) is less than 17.47 m^3 (617 ft^3), Figure 15.06.05-52-1 shows a liquid inventory of 21.2 m^3 (750 ft^3) to 25.5 m^3 (900 ft^3) at 6,000 sec, which exceeds the value used in the original analysis in ANP-10288P, Revision 0. Based on the time-dependent results for the concentrating region liquid inventory from the analyzed three cases, the applicant concluded that the original assumption was conservative. The staff considered this conclusion reasonable as the presented S-RELAP5 results clearly showed that the available liquid inventory in the concentrating region, predicted at 6,000 sec, exceeds by a large amount the value of 17.47 m^3 (617 ft^3). Although this volume was applied in the boron precipitation analysis from the very beginning, the fact that S-RELAP5 analysis predicted a smaller liquid volume quite early into the transient as seen in Figure 15.06.05-52-1, "Water Inventory in the Five Sub Volume Core Concentrating Region," during the initial 1,000 sec or so, the boron concentration expected that early into the transient would be significantly below the precipitation limit. Nonetheless, the staff further questioned additional aspects related to the calculation of the concentrating region liquid content. Therefore, in follow-up RAI 403, Questions 15.06.05-64, 15.06.05-65, and 15.06.05-67, the staff requested that the applicant clarify additional aspects related to the calculation of the concentrating region liquid content as discussed below.

In RAI 241, Question 15.06.05-53, the staff requested that the applicant provide a description for the calculation of the initial boron concentrations within the control mixing volume for the U.S. EPR post-LOCA boric acid accumulation analysis. In a November 25, 2009, response to RAI 241, Question 15.06.05-53, the applicant stated that the initial concentration in the mixing volume for the boron precipitation calculation was set equal to the maximum injected boron concentration, which was determined by combining the available sources of borated water. These sources included two EBS water storage tanks, four accumulators, the IRWST, and the RCS. The applicant determined maximum injected boron concentrations, as reported in the November 25, 2009, response to RAI 241, Question 15.06.05-53, was 1,929 ppm, 37 percent enriched boron. The staff considered this value to be slightly higher than the value of 1,925 ppm calculated in ANP-10288P, Revision 0. Although the discrepancy is negligible, the staff addresses it later in this section. Furthermore, the response provided assessment results for the initial boron concentrations based on a simple thermodynamic expression for the quality of the expanded fluid to calculate the fraction of water that evaporates during the LOCA initial blowdown process. Three LOCA cases with the same initial RCS conditions and different end-of-blowdown pressures (414 kPa (60 psia), 310 kPa (45 psia), and 101.3 kPa (14.7 psia)) were analyzed. An RCS average initial coolant temperature of 312.4 °C (594.4 °F) and an RCS initial pressure of 15.51 MPa (2,250 psia) were applied in the assessment. These values match the average values for the temperature and pressure sampling ranges in the U.S. EPR realistic LBLOCA analysis in FSAR Tier 2, Table 15.6-8, "RLBLOCA - Sampled Plant Parameters (Cycle 1 and Equilibrium Cycle)," and agree with the values for the SBLOCA analysis in FSAR Tier 2, Table 15.6-14, "SBLOCA - U.S. EPR System Analyses Parameters." A maximum 18-month cycle RCS operating boron concentration of 855 ppm was assumed in the assessment. The calculations used a simple equilibrium thermodynamic relationship to calculate the quality of the expanded fluid using the average enthalpy of the water circulating in the RCS just prior to the LOCA. The calculated results for the post-LOCA initial boron concentrations were provided in Table 15.06.05-53-1 of the November 25, 2009, response to RAI 241, Question 15.06.05-53. The maximum calculated initial boron concentration value of 1,521 ppm corresponded to the case with the minimum assumed end-of-blowdown pressure of 101.3 kPa (14.7 psia). Based on this outcome, the applicant concluded that the use of 1,929 ppm as the initial concentration was conservative.

In view of the response to RAI 241, Question 15.06.05-53, the staff concludes that using the latent heats of evaporation based on the assumed end-of-blowdown pressures led to non-conservative results due to the strong dependency of the latent heat of evaporation on the saturation pressure. Using the same initial conditions and a latent heat of evaporation based on the average of the initial RCS water temperature of 312.4 °C (594.4 °F) and the end-of-blowdown temperature of 100 °C (212 °F) at 101.3 kPa (14.7 psia), the staff calculated a post-LOCA initial boron concentration of 1,918 ppm with the maximum 18-month cycle RCS operating boron concentration of 855 ppm, a result that is very close to the assumed value of 1,929 ppm in the November 25, 2009, response to RAI 241, Question 15.06.05-53. The staff also notes that the impact of heat generation in the core was not considered in the above approach nor was the contribution of heat release from the RPV or internal metal structures. In the case of an SBLOCA, core steaming that takes place prior to the point at which the RCS pressure falls below the ECCS injection pressure, will also have an impact on the initial boron concentration. The staff concluded that justification for the limiting reactor coolant system operating boron concentration value used in the response to RAI 241, Question 15.06.05-53, was necessary. The staff will review such additional information in conjunction with RAI 403, Question 15.06.05-68, which is currently tracked as an open item and is discussed below.

In RAI 241, Question 15.06.05-54, the staff requested that the applicant justify the applicability of an average boron concentration, calculated over a control mixing volume comprising the U.S. EPR core fluid volume, parts of the upper and lower plena, and the fluid volume in the guide tubes and heavy reflector, as a criterion for ensuring boron precipitation prevention. In a November 25, 2009, response to RAI 241, Question 15.06.05-54, the applicant stated that as the boiling coolant flows up through core regions, the liquid boron concentration gradually increases and reaches its maximum at the top of the core. The applicant explained that boron in the water in the upper plenum and in the downward flow returning to the lower plenum through the guide tubes and the heavy reflector holes upon the establishment of a recirculation flow pattern involving these regions and the core is at this maximum concentration. The applicant additionally stated that the only region where the concentration could be increased above the upper plenum concentration is in the peripheral fuel assemblies where a small amount of steam is generated as water flows downwards to the lower plenum as discussed in the staff's evaluation of RAI 241, Question 15.06.05-55, below. The applicant stated that the average concentration corresponds to the actual concentration somewhere along the core flow path above the point where boiling starts and the core exit. Thus, the average boron concentration over-predicts the concentration in the core region taken as a whole, as well as the actual concentration in some areas of the core. The applicant concluded that using an average boron concentration is a reasonable simplification considering that there is enough predicted margin to the precipitation limit as discussed in the analysis in support of the response to RAI 241, Question 15.06.05-55, reviewed below. The staff finds the applicant's conclusions acceptable because the analysis results, based on the applicant's assumptions, conservatively over-predict the calculated boron concentration in the core region as a whole while preserving an ample margin to the precipitation limit according to the analysis results presented in the response to RAI 241, Question 15.06.05-55.

In RAI 241, Question 15.06.05-55, the staff requested that the applicant justify the inclusion of liquid-containing regions located outside of the active core in the mixing (control) volume. The staff also requested a description of the mechanisms leading to liquid mixing in the core and in the remaining sub-volumes included in the mass balance control volume. In a November 25, 2009, response to RAI 241, Question 15.06.05-55, the applicant stated that in addition to the single, constant volume simplification, several additional conservative assumptions were made in the boron precipitation modeling approach. The assumptions included: All of the boron in the water that was evaporated was deposited to the region so that there was no carry out of boric acid due to droplet entrainment or steam solubility; no increased boron solubility due to other solutes; no increased boiling temperature due to containment pressure and boric acid concentration; and no addition of water containing less boric acid from sources such as the CVCS. In the response, the applicant also referred to the S-RELAP5

RLBLOCA analysis of the 18-month equilibrium cycle Case 24 run, which was extended to 6,000 sec, which was also discussed in the November 25, 2009, response to RAI 241, Question 15.06.05-52. In this case, the recirculation pattern involving the guide tubes, the heavy reflector, and the peripheral core region was predicted to occur at approximately 1,000 sec. The applicant stated that the steam flow entering the upper plenum was high enough to thoroughly mix the upper plenum region. Therefore, the applicant concluded that the inclusion of the upper plenum in the concentrating volume was appropriate.

In the November 25, 2009, response to RAI 241, Question 15.06.05-55, the applicant also provided results from a sensitivity study performed with a two-region boron precipitation model. In this model, the lower plenum in the original model was separated as a stand-alone region. The second region, referred to as a core region, comprised the remaining four sub-volumes in the original concentrating volume. In the sensitivity calculation performed with the two-region boron precipitation model, both regions were coupled at 1,000 sec in agreement with the S-RELAP5 prediction for the LBLOCA case. As in the original single-volume precipitation model presented in ANP-10288, the lower plenum and the core region had an initial boron concentration of 1,929 ppm. The staff notes that the results showed that the core region accumulated a higher boron content in comparison to the boron concentration predicted for the lower plenum. In addition, the two-region model predicted that the core region would reach the 100 °C (212 °F) mixing precipitation limit at a later point in time when compared to the single, constant volume model. Based on the comparative results, the applicant stated that the original analysis was conservative and adequate for determining the bounding time to switch to simultaneous injection to prevent boron precipitation. Although insightful, the staff concluded that the application of a two-region boron precipitation model by itself cannot be used to justify the acceptability of the results obtained with the original single-region concentrating model, as both approaches have the same modeling limitations and inherent uncertainties. At the same time, given the prediction of a recirculation pattern involving all five sub-volumes (the heated core region, heavy reflector, guide tube regions, lower support plate to heated core, and upper plenum up to the bottom of the hot legs) with the U.S. EPR realistic LBLOCA S-RELAP5 model, the staff agrees that their inclusion in the concentrating mixing volume was acceptable.
In particular, the staff notes that this recirculation pattern takes place after core quench following the LBLOCA. After core quench, denser liquid accumulates over the heavy reflector and the peripheral fuel assemblies and flows downward through the low power peripheral fuel assemblies, heavy reflector, and guide tubes into the lower plenum. In the lower plenum, this liquid mixes with the ECCS flow and returns upwards to the inner core regions. Additionally, the steam flow rate entering the upper plenum is much greater than the liquid flow rate so that it thoroughly mixes the region. Accordingly, the staff concludes that it is acceptable for the upper plenum to be also included in the core concentrating region. The staff also considered sensitivity results from a supplementary two-region precipitation model presented in the November 25, 2009, response to RAI 241, Question 15.06.05-55. These results illustrated that the single-region, constant-volume model yielded more restrictive results as it predicted a time period to precipitation of 2.2 hours compared to 4.3 hours from the two-region model. Accordingly, the staff finds the applicant’s conclusions acceptable because the applicant’s overall assumptions are conservative with respect to calculation of boron concentration using the proposed single-region mixing volume. Therefore, the staff considers RAI 241, Questions 15.06.05-54 and 15.06.05-55, resolved.

The applicant submitted ANP-10288P, Revision 1, in January 2010. FSAR Tier 2, Section 15.6.5 cites this revision of ANP-10288P and summarizes the results documented in more detail in the revised ANP-10288P technical report. The staff reviewed the updated information presented in ANP-10288P, Revision 1, and noted that the initial concentration of 1,925 ppm as calculated in ANP-10288P, Revision 0, was revised to 1,929 ppm, which is in agreement with the value provided in the response to RAI 241, Question 15.06.05-53, as discussed above. Both values are very close and can be considered consistent. Also, the staff notes that the revised result is slightly more conservative for the precipitation analysis as it is somewhat higher. Nonetheless, the staff examined the reason for this deviation and determined that the discrepancy was due to using a combined volume of 73.3 m^3 (2,590 ft^3) for both EBS storage tanks at their maximum volume in ANP-10288P, Revision 1, instead of the 71.5 m^3 (2,526 ft^3) previously used in ANP-10288P, Revision 0. Although the last combined EBS volume agreed more closely with the volume of 71.5 m^3 (2,527 ft^3) provided in FSAR Tier 2, Table 6.8-1, the staff finds that the revised initial concentration of 1,929 ppm is slightly more conservative. In addition, the volumes of the five sub-regions comprising the liquid volume of the concentrating region for both the small-break and large-break LOCA categories were somewhat corrected and so were the voided fraction values associated with each of the sub-regions as seen from Tables 2-2 and 2-3 in ANP-10288P, Revision 1, and in ANP-10288P, Revision 0. In ANP-10288P, Revision 1, the void fraction predictions were based on S-RELAP5 results for the new limiting PCT SBLOCA break size of 16.51 cm (6.5 in.) diameter at 1,700 sec and the 0.437 m^2 (4.7 ft^2) limiting large-break size from the RLBLOCA analyses. This LBLOCA case was different from the 0.381 m^2 (4.1 ft^2) break case analysis used to evaluate the effectiveness of the hot leg injection in both ANP-10288P, Revision 1, and ANP-10288P, Revision 0. The control volume liquid content values provided in Tables 2-3 and 2-3 in ANP-10288P, Revision 1, for both the small-break and large-break LOCA categories were less than the values found in Tables 2-3 and 2-3 in ANP-10288P, Revision 0. The SBLOCA analyses, performed with the applicant's S-RELAP5-based methodology, assumed automatic partial cooldown at 82.2 °C/hr (180 °F/hr) followed by manual initiation of cooldown at 32.2 °C (90 °F/hr) starting at 1,800 sec. ANP-10288P, Revision 1, stated that for break sizes up to 10.16 cm (4 in.) diameter, two trains of ECCS can refill the RCS and establish natural circulation within 4 hours thus limiting the period of pool boiling to less than 3 hours. For the 5.08 cm (2 in.) break case, the natural circulation stopped at 1,500 sec and two-phase circulation started at 7,500 sec followed by single-phase flow established at 10,000 sec. Core outlet void fraction increased to 40 percent in 1,500 sec and then slowly increased to 45 percent at 7,500 sec, at which time the LHSI started to refill the core and active coolant loops. In the 16.51 cm (6.5 in.) limiting PCT SBLOCA case presented in ANP-10288P, Revision 1, the redirection of LHSI flow to the hot leg results in a hot leg injection flow pattern that returns the LHSI flow from the loops receiving hot leg injection to the upper plenum and down the peripheral regions of the core. This upper plenum and core flow behavior was found similar to that observed in test facilities with hot leg injection such as the previously-mentioned SCTF, CCTF, and the UPTF. This analysis showed that hot leg injection was effective at penetrating the core, thus reducing the boron concentration in the core region and preventing precipitation of boron in the core and RCS. ANP-10288P, Revision 1, stated that small breaks larger than about 16.51 cm (6.5 in.) diameter are too large to allow two trains of LHSI and MHSI to refill the loops and establish natural circulation. Similar to LBLOCAs, the core remains in a pool boiling mode; however, the water content in the core concentrating region is larger than the LBLOCA and as such the larger SBLOCAs are bounded by the evaluation of LBLOCA for boron precipitation. The LBLOCA case presented in ANP-10288P, Revision 1, was the same as that described in ANP-10288P, Revision 0, and discussed above.

Based on the evaluation of the responses to RAI 241, Questions 15.06.05-52 through 15.06.05-55, the updated information provided in ANP-10288P, Revision 1, information obtained at the onsite reactor systems audit held in Lynchburg, VA on November 18-19, 2009, and at a public meeting on July 15, 2010, the staff issued specific follow-up questions on boron precipitation in RAI 403, Questions 15.06.05-64 through 15.06.05-68, were issued on June 2, 2010, as discussed below.

In RAI 403, Question 15.06.05-64, the staff requested that the applicant provide an analysis of boric acid build-up with a break on the top of the cold leg pipe and the loop seal vertical section completely filled with two-phase fluid. In a May 31, 2011, response to RAI 403, Question 15.06.05-64, the applicant stated that a constant concentrating volume extracted from the RLBLOCA transient and then held constant throughout the boron precipitation analysis may not be conservative. To evaluate the potential impact of associated refilling loop seals with a top of pipe break and decrease in the core concentrating volume, the applicant applied the long-term core cooling static-balance model, addressed in a draft response to RAI 403, Questions 15.06.05-69 through 15.06.05-77, to obtain an estimate of reactor vessel water inventories during the early post-reflood period. The results obtained by the applicant revealed that using the time-dependent concentrating volumes from the static balance model increased the time to boron precipitation by 1.5 hrs. Therefore, the staff agrees with the response conclusion that using a constant volume from the end of the RLBLOCA transient is conservative for the analysis of the time to boron precipitation and finds the response acceptable. Accordingly, the staff considers RAI 403, Question 15.06.05-64, resolved.

In RAI 403, Question 15.06.05-65, the staff requested that the applicant provide results showing the sensitivity of the timing for boric acid precipitation to the applied axial power shape. In a May 31, 2011, response to RAI 403, Question 15.06.05-65, the applicant agreed that a bottom peaked axial shape could result in more vapor within the core region, which would decrease the concentrating region liquid content. To address this issue, the applicant reanalyzed a base S-RELAP5 case representing the smallest end-of-RLBLOCA-transient volume from the Cycle 1 RLBLOCA analysis using the ANP-10278P, Revision 0, methodology. For the sensitivity case, the bottom peaked axial shape was replaced with a top peaked axial shape. The results showed that the liquid volume in the active core with the top peaked case was approximately 1.982 m^3 (70 ft^3) greater than for the bottom peaked case. In addition, the applicant presented a similar comparison of two additional sensitivity cases evaluated with the static-balance model and discussed in the response to RAI 403, Question 15.06.05-64. The increase in the predicted liquid volume using the static-balance model was approximately 1.416 m^3 (50 ft^3). While admitting that there was an effect of the axial shape on the time-dependent concentrating volume, the applicant presented a plot comparing the concentrating region volumes from both S-RELAP5 cases against the value used in the end-of-RLBLOCA-transient constant volume approach used in the boron precipitation analysis. The comparison showed that the concentrating region volumes from both S-RELAP5 cases quickly exceed the end-of-RLBLOCA-transient constant volume thus bounding the small increase resulting from a bottom peaked axial shape. Based on these results of sensitivity studies performed by the applicant to address the effect of a bottom peaked axial shape, the staff concludes that the conservatism associated with the constant volume approach in the U.S. EPR boron precipitation analysis bounds the effects associated with the axial power shape profile. Therefore, the staff agrees with the applicant's conclusion that the U.S. EPR boron precipitation analysis is conservative and finds the response acceptable. Accordingly, the staff considers RAI 403, Question 15.06.05-65, resolved.

In RAI 403, Question 15.06.05-66, the staff requested that the applicant describe the entrainment model and show the results demonstrating that the hot leg ECCS injection is effective in preventing boron precipitation. The staff also requested that the applicant show the applicability of the Wallis entrainment correlation with regard to the conditions under which it was applied in the analysis. **RAI 403, Question 15.06.05-66, is being tracked as an open item.**

In RAI 403, Question 15.06.05-67, the staff requested that the applicant validate the S-RELAP5 level swell model against low pressure level swell data such as the data obtained from the ACHILLES and THETIS low pressure level swell tests. In a May 31, 2011, response to RAI 403, Question 15.06.05-67, the applicant presented results from a study to assess S-RELAP5 capabilities for predicting core level swell at low pressure conditions. The assessment study was based on Test A1L066, which is part of the low pressure level swell experiments performed at the ACHILLES test rig. Test A1L066 was performed at an absolute pressure of 1.2 bar. For the purposes of the assessment study, an S-RELAP5 model representing only the fuel bundle test section of the entire rig was developed. A TWODEE hydrodynamic component with inner and peripheral regions was used to model the test section. Heat structures were coupled with the TWODEE component to model the electrically heated rods in both regions. In addition, heat structures were used to model the shroud heating of the peripheral region as a thin-wall tube. The downcomer and the inlet connection piping were not modeled and instead the test section inlet volumetric flow rate and water temperature were defined as inlet flow boundary conditions in the model. The test pressure was defined as an exit boundary condition, and a constant power was applied to the heater rods and to the shroud heaters. As the model represents the heated test section of the rig and captures the test conditions determining the level swell response, the staff finds it acceptable for producing level swell calculations for the purpose of benchmarking the code at low pressures. The results from this benchmarking study were presented in the May 31, 2011, response to RAI 403, Question 15.06.05-67, and reviewed by the staff with the following findings.

The prediction results obtained with S-RELAP5 without modifying the code two-phase flow models were compared against Test A1L066 measurements for the test bundle mixture level and the collapsed liquid level. For the purpose of the comparison, the applicant used two different techniques for quantifying the two-phase mixture level based on the computed results. In the first technique, the mixture level was defined as the elevation at the top of the node where the heater rod surface temperature changed from near saturation temperature to an increase of 5 °C (9 °F) higher in the node above. In the second approach, the mixture level was determined as the sum of node lengths to the elevation at the top of the node where the void fraction above was greater than 95 percent. Alternatively, an extrapolation technique using the predicted void fraction in the top fully covered node just below the node containing the mixture level could have been used. The applicant provided comparison plots for the two-phase mixture levels predicted in the central and peripheral bundle regions again test data With the exception of observed under-prediction of the mixture levels relative to the ACHILLES data prior to 100 s of the test, the S-RELAP5 results compared favorably with the data during the remaining part of the transient. Based on the presented validation results, the staff concludes that S-RELAP5 predicts reasonably well low-pressure level-swell. The staff finds the May 31, 2011, response to RAI 403, Question 15.06.05-67, acceptable. Accordingly, the staff considers RAI 403, Question 15.06.05-67, resolved.

In RAI 403, Question 15.06.05-68, the staff requested that the applicant present the results for the timing to boric acid precipitation when the concentration of the fluid entering the core is based on individual flow rates and concentrations characterizing various sources of injection into the RCS. **RAI 403, Question 15.06.05-68, is being tracked as an open item.**

15.6.5.3.4.4 *In-Vessel Downstream Effects of Containment Debris*

The staff's safety evaluation of the in-vessel downstream effects of containment debris, including the review of technical report ANP-10293P, Appendix F, "Downstream Effects Evaluation for the U.S. EPR," will be provided at a later date.

15.6.5.3.5 Combined License Information Items

There are no COL information items related to this area of review. The staff determined that no COL information items need to be included in FSAR Tier 2, Table 1.8-2, "U.S. EPR Combined License Information Items," for post-LOCA long-term cooling consideration.

15.6.5.3.6 Conclusions

The staff's conclusions will be made at a later date upon resolution of the open items enumerated above.