15 TRANSIENT AND ACCIDENT ANALYSES

15.1 Introduction

In Chapter 15 of the AP1000 Design Control Document (DCD), the applicant discusses the analysis of various design-basis transients and accidents. The applicant uses the results of these analyses in the DCD to show the conformance of the AP1000 advanced passive plant design with general design criterion (GDC) 10 for fuel design limits, GDC 15 for the reactor coolant pressure boundary (RCPB) pressure limits, and the requirements of Title 10 of the Code of Federal Regulations (10 CFR) 50.46 for the performance of the emergency core cooling system (ECCS).

The staff has reviewed the AP1000 transient and accident analyses in the DCD, in accordance with Chapter 15 of the NRC's Standard Review Plan (SRP), NUREG-0800.

15.1.1 Event Categorization

The applicant assigns the initiating events to the following categories in accordance with Chapter 15 of the SRP and Regulatory Guide (RG) 1.70, "Standard Format and Content of Safety Analysis Reports for Nuclear Plants:"

- increase in heat removal from the primary system

- decrease in heat removal by the secondary system

- decrease in reactor coolant system (RCS) flow rate

- reactivity and power distribution anomalies

- increase in reactor coolant inventory

- decrease in reactor coolant inventory

- anticipated transients without scram (ATWS)

The first category, an increase in heat removal from the primary system, includes a new event involving inadvertent operation of the passive residual heat removal (PRHR) heat exchanger (HX). Because the category is a broader categorization than the SRP and RG 1.70 category of increase in heat removal by the secondary system, and reflects the AP1000 design, it is acceptable.

The applicant also groups the design-basis events according to their anticipated frequency of occurrence identified as Condition I - normal operation and operational transients; Condition II - faults of moderate frequency; Condition III - infrequent faults; and Conditions IV- limiting faults.

The applicant's event frequency grouping is consistent with the guidelines of RG 1.70 and the criteria of American Nuclear Society (ANS) 18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants." Condition I events occur frequently and should be considered from the point of view of their effects on the consequences of Conditions II, III, and IV events. Condition II events are those that may occur during a calendar year for a particular plant. Condition III events are those that may occur infrequently during the life of a particular plant. Condition IV events are postulated but not expected to occur during the life of a plant.

The SRP divides the events into anticipated operational occurrences (AOOs) and postulated accidents. The requirements of 10 CFR Part 50, Appendix A, define AOOs as conditions of normal operation and those transients that are expected to occur one or more times during the life of a plant; therefore, AOOs encompass the normal, moderate frequent and infrequent events of Conditions I through III. Chapter 15 of the SRP does not specify a category of infrequent incidents but does provide specific acceptance criteria for those events that cannot be categorized as infrequent. Thus, the event frequency categorization of the DCD is consistent with the NRC's licensing approach.

In Section 15.0.1 of the DCD, the applicant lists the design-basis events analyzed under Conditions II, III, and IV. These events are generally consistent with the current licensing practice. However, the complete loss of forced reactor coolant flow event and single rod cluster control assembly (RCCA) withdrawal event at full-power condition are listed in the DCD as Condition III - infrequent faults. The event categorization is inconsistent with SRP 15.3.1 and 15.4.3 that classify complete loss of RCS flow and the withdrawal of a RCCA as Condition II - moderate frequent events with the acceptance criteria that specify no violation against the safety limit of the departure from nucleate boiling ratio (DNBR). Nonetheless, the applicant analyzes the complete loss of forced reactor coolant flow event as presented in DCD Section 15.3.2 to satisfy the acceptance criteria for a Condition II event. Thus, the staff concludes that the applicant's approach for the analysis of the complete loss of reactor coolant flow event is acceptable.

The withdrawal of a RCCA event was previously allowed (WCAP-9272-A, "Westinghouse Reload Safety Evaluation Methodology") to be classified as a Condition III event for reactors manufactured by Westinghouse because of its very low probability of occurrence. Since the AP1000 reactor is designed by the same manufacturer for the existing pressurized-water reactors (PWRs), the applicant's classification of the withdrawal of a RCCA event as a Condition III event is consistent with the current licensing practices, and therefore, is acceptable.

15.1.2 Non-Safety-Related Systems Assumed in the Analysis

For the design-basis analysis, only safety-related systems or components are allowed to be used to mitigate the events. In Westinghouse letter ET-NRC-93-3804, dated January 22, 1993, the applicant's response to the staff's request for additional information (RAI) 440.31 for the AP600 review stated that non-safety-related systems or components are assumed to be operational in the following situations:

(a) when assumption of a non-safety-related system results in a more limiting transient,

(b) when a detectable and non-consequential random, independent failure must occur in order to disable the system, and

(c) when non-safety-related components are used as backup protection.

For the AP1000 design, the applicant applies, as indicated in DCD Tier 2 Chapter 15, the same approach as the AP600 design in using non-safety-related systems or components for the analyses of the design-basis events.

Case (a) is an assumption that will result in a more limiting transient, therefore, it is acceptable.

For Case (b), the applicant assumes continued operation of the main feedwater control system (MFCS) in the design-basis analysis of those events not related to feedwater system malfunction, loss of alternating current (ac) power, or turbine trip. For example, an event involving withdrawal of a rod cluster control assembly (RCCA) is analyzed from an at-power condition. Before the initiating fault causing the RCCA withdrawal, the MFCS should be operating and maintaining steam generator inventory. If a failure exists in the MFCS, it should be detectable in the control room by alarms or abnormal control system performance before the start of the RCCA withdrawal event. The staff concludes that the assumption of MFCS continued operation is acceptable since a failure in the MFCS is not a consequence of the initiating event, and the probability of a random, independent failure occurring in the MFCS within the time frame of the initiating event is extremely low.

For Case (c) as discussed in the response to RAI 440.061 and summarized in DCD Table 15.0-8, the applicant credits the following non-safety-related components as backup protection in the design-basis analysis for the AP1000 design:

- The main feedwater pump trip in the analysis of an increased feedwater flow event.
- The pressurizer heater block in the analysis of loss of normal feedwater, inadvertent operation of core makeup tanks (CMTs), chemical and volume control system (CVCS) malfunction that increases reactor coolant inventory, steam generator tube rupture (SGTR), and small-break loss-of-coolant accidents (SBLOCA).
- Main steam isolation valve (MSIV) backup valves, and main steam branch isolation valves in the analysis of inadvertent opening of steam generator (SG) safety valves, steamline break (SLB), and SGTR events. The MSIV backup valves include the turbine stop, control valves, the turbine bypass valves, and the moisture separator reheat steam supply control valve.

During the course of the review, the staff asked the applicant to address its compliance with 10 CFR 50.36, which specifies the criteria for the systems that are subject to technical specification (TS) limiting conditions for operation (LCOs). Specifically, 10 CFR 50.36(c)(2)(ii)(C) requires, in part, that a TS be established for a structure, system, or

component that is assumed to function or actuate in a design-basis analysis for mitigation of specified events. In its response to RAI 440.061, the applicant indicates that it complies with the 10 CFR 50.36 requirements by providing TSs to include non-safety-related systems that are credited as backup systems in the licensing design-basis analyses. Items 7 and 27 of TS Table 3.3.2-1 include applicable modes, surveillance requirements and trip setpoints for the main feedwater pump trip and pressurizer heater trip, respectively. The LCOs for the main steam branch isolation valves and the MSIV backup valves are provided in Section 3.7.2 of the TS. These TSs ensure the reliability of the non-safety related components credited as backup systems in the design-basis analyses. Therefore, Section 3.7.2 of the AP1000 TS is acceptable.

Based on its review, the staff concludes that crediting these non-safety-related backup protection systems and components in the design-basis analyses is acceptable for the following reasons:

- The trip mechanisms of the feedwater pump trip breakers and pressurizer heater trip breakers are simple, and the likelihood of the breaker function failure is low.

- The operating data show that the turbine stop and control valves are reliable, and taking credit of the turbine valves in the design-basis analyses for backup protection is consistent with the staff position stated in NUREG-0138, "Staff Decision of Fifteen Technical Issues Listed in Attachment to November 3, 1976, Memorandum from Director, NRR to NRR Staff."

- The applicant has included surveillance requirements and limiting conditions for operation in the TSs to ensure the reliability of the following systems or components:

  (1) feedwater pump trip breakers and redundant pressurizer heater trip breakers, and
  (2) the MSIV backup valves and the main steam branch isolation valves.

15.1.3 Chapter 15 Loss of Offsite Power Assumptions

As indicated in DCD Tier 2 Section 15.0.14 of the DCD, the applicant performs the Chapter 15 analysis assuming a loss of offsite ac power (LOOP). The LOOP is not considered as a single failure, and the analysis is performed without changing the event category. The assumption of the LOOP in the Chapter 15 analysis complies with the requirements of GDC 17 of 10 CFR Part 50, Appendix A, which requires anticipated operational occurrences and postulated accidents to be analyzed assuming a LOOP. In the analysis, a LOOP is considered a consequence of an event as a result of disruption of the grid following a turbine trip during the event.

In the case of events involving a turbine trip, the applicant assumes that a LOOP and the resulting coastdown of the reactor coolant pumps occurs 3 seconds after the turbine trip. The basis for the 3-second delay is provided in DCD Tier 2 Section 8.2. That section describes the electrical design features of the AP1000, the electrical system response to a turbine trip, and the combined license (COL) applicant interfaces that support the 3-second assumption. Among

others, the AP1000 design provisions include the following electrical features that support the 3-second delay:

- Use of an output generator circuit breaker and reverse power relay with at least a 15-second delay before tripping the breaker following a turbine trip (this allows the generator to provide voltage support to the grid and maintain adequate voltage to the reactor coolant pumps (RCPs) for significantly longer than the assumed 3-seconds).

- The COL applicant interface item in item 8.3 DCD Tier 2 Table 1.8-1, Item 8.3, that transient stability must be maintained and the RCP bus voltage must remain above the voltage required to maintain the flow assumed in Chapter 15 analyses for a minimum of 3 seconds following a turbine trip (this ensures that, for the applicant's unique grid system configuration, a grid instability condition following a turbine trip will take at least 3 seconds before it results in a loss of power to the RCPs).

- The COL applicant interface item in DCD Tier 2 Table 1.8-1 that the protective devices controlling the switchyard breakers are set with consideration for preserving the plant grid connection following a turbine trip (this is especially important in generator output circuit breaker designs to ensure that the backfeed offsite circuit through the generator main stepup transformer is not interrupted by opening of the switchyard breakers following a turbine trip).

- No automatic transfers of RCP buses are used in the design (this precludes bus transfer failures following a turbine trip).

- If a turbine trip occurs when the grid is not connected to the plant, the main generator will be available to power the RCPs for at least 3 seconds before the generator output breaker is tripped on generator under-voltage or exciter over-current.

The staff has reviewed the information on the AP1000 electrical design, as well as the COL requirements. On that basis, and as described above, the staff has reasonable assurance that the RCPs can receive power for a minimum of 3 seconds following a turbine trip (see Section 8.2.3.6 of this report). The staff has also reviewed the DCD Tier 2 Chapter 15 analysis and found that the applicant considers LOOP in all of the applicable analyzed events and applies the acceptance criteria specified in the related SRP sections for events with and without LOOP. Therefore, the staff concludes that the applicant's approach is acceptable.

15.1.4 Analytical Methods

The analytical methods used for transient and accident analyses are normally reviewed on a generic basis. As indicated in DCD Tier 2 Sections 15.0.11, 15.6.5.4A and 15.6.5.4B of the DCD, the methods for transient and accident analyses include the following computer codes:

- TWINKLE: This multi-dimensional spatial neutronic code uses an implicit finite-difference method to solve the two-group transient neutronic equations in one, two, and three dimensions. TWINKLE has been used to calculate the kinetic response of a

reactor for transients, such as the RCCA bank withdrawal from subcritical conditions and RCCA ejection events, which cause a major perturbation in the spatial neutron flux distribution. As documented in WCAP-7979-P-A (Proprietary) and WCAP-8028-A (Non-Proprietary) dated January 1975, this code has been approved by the NRC for operating Westinghouse plants and the AP600. Since the AP1000 fuel design is similar to that of operating Westinghouse plants and the AP600, i.e., falls within the NRC-approved applicable range of the code, the application of the TWINKLE code to the AP1000 for analysis of kinetic responses is acceptable.

- VIPRE-01: As documented in WCAP-14565-P-A (Proprietary) and WCAP-15306-NP-A (Non-Proprietary), the NRC has approved this code for the core thermal-hydraulic analyses, determining coolant density, mass velocity, enthalpy, vapor void, static pressure, and the DNBR distribution along parallel flow channels within the reactor core under normal operational and transient conditions. Since the AP1000 core design is similar to that of operating Westinghouse plants and the AP600, i.e., falls within the NRC-approved applicable range of the code, the application of the VIPRE-01 code to the AP1000 thermal-hydraulic calculations is acceptable.

- COAST: As documented in CENPD-98-A, the NRC has approved this code for use in calculating the reactor coolant flow coast transient for any combination of active and inactive reactor coolant pumps and forward and reverse flow in the hot- or clod-legs. The code is approved for the PWRs manufactured by the former Combustion Engineering company (now merged as part of Westinghouse Company.) In the COAST code, the equations of conservation of momentum are formulated for each of the flow paths of the COAST model assuming unsteady state one-dimension flow of an incompressible fluid. The equation of conservation of mass is formulated for each nodal point. Pressure losses resulting from friction and geometric losses are assumed proportional to the flow velocity square. RCP dynamics are modeled using a head-flow curve for a pump at full-speed and using four-quadrant curves, which are parametric diagrams of pump head and torque on coordinates of speed versus flow, for a pump at other than full-speed. The COAST code is a generic code for calculating the coastdown flow with the values of pump head curves and the pressure drop coefficients of the RCS components as the input parameters. Since the pump head curves and pressure loss coefficients used in the COAST code reflect the AP1000 design, the staff concludes that the use of COAST is acceptable for the AP1000 in calculating the RCP flow during the coast transient.

- FACTRAN: As documented in WCAP-7908-A (proprietary) and WCAP-7337-A (non-proprietary), the NRC has approved the FACTRAN code for calculations of the transient heat flux at the surface of a rod. Since the AP1000 fuel rod design is similar to that of operating Westinghouse plants and the AP600, i.e., falls within the NRC-approved applicable range of the code, the application of FACTRAN to the AP1000 heat flux calculations is acceptable.

- LOFTRAN: As documented in WCAP-7907-P-A (proprietary) and WCAP-7907-A (non-proprietary), the NRC previously approved this code to allow Westinghouse to analyze

system responses to non-LOCA events for conventional Westinghouse PWRs. LOFTRAN simulates a multi-loop system using a model containing a reactor vessel, hot and cold leg piping, steam generators, and pressurizer. The code also includes point kinetics and reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the steam generator uses a homogeneous, saturated mixture for analyses of thermal transients and a water level correlation for indication and control. When the applicant applied the LOFTRAN code to the AP600 safety analysis, it modified the code to incorporate features representative of the AP600 design which are important to modeling the non-LOCA transient analyses. The LOFTRAN modifications are documented in WCAP-14234, “LOFTRAN and LOFTTR2 AP600 Applicability Document,” and supported by test data comparisons as documented WCAP-14307, “AP600 LOFTRAN-AP and LOFTTR2-AP Final Verification and Validation Report.”

- LOFTTR2: As documented in WCAP-10698-P-A (proprietary) and WCAP-10750-A (non-proprietary) dated August 1985, and supplemented by WCAP-10759-A and WCAP-11002, the NRC-approved code is used to analyze an SGTR event for conventional Westinghouse PWRs. LOFTTR2 is a modified version of LOFTRAN with a more realistic break flow model, a two-region steam generator secondary side, and an improved capability to simulate operator actions during an SGTR event. The version of LOFTTR2 applied to the AP600 SGTR analysis incorporated the LOFTRAN changes to simulate passive safety features for the AP600 design. These changes are documented in WCAP-14234.

- NOTRUMP: This code consists of the modeling features that meet the requirements of Appendix K to 10 CFR Part 50. As documented in WCAP-10079-P-A and WCAP-10054-P-A dated August 1985, the NRC previously approved the NOTRUMP code for the SBLOCA analysis. The modified version of the NOTRUMP code for the AP600 application is documented in WCAP-14807, “NOTRUMP Final Verification and Validation Report.”

- WCOBRA/TRAC-LBLOCA: As documented in WCAP-12945, this “best estimate” code has been approved by the NRC in a safety evaluation dated June 28, 1996, for large-break loss-of-coolant accident (LBLOCA) analysis. The modified version of the WCOBRA/TRAC code for the AP600 application is documented in WCAP-14171, “WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident.” Its auxiliary code, HOTSPOT, is updated for the AP600 and documented in Westinghouse letter NSD-NRC-97-5171 dated June 10, 1997.

- WCOBRA/TRAC-LTC and WGOTHIC: WCOBRA/TRAC is also used for the post-LOCA long-term cooling (LTC) analyses. The code verification for the long-term cooling analyses is documented in WCAP-14776, “WCOBRA/TRAC, OSU Long-Term Cooling Final Validation Report.” The WGOTHIC code, documented in WCAP-14407, “WGOTHIC Application to AP600,” is used to calculate containment boundary conditions for LBLOCA and post-LOCA long-term cooling. The staff previously reviewed and accepted the application of WCOBRA/TRAC and WGOTHIC for long-term cooling calculations. This is discussed in Chapter 21 of this report.

In support of the AP1000 application, the applicant submitted a topical report WCAP-15644, "AP1000 Code Applicability Report," for the staff to review. WCAP-15644 documents the applicant assessment of the safety analysis codes that were developed and approved for the AP600 design certification to determine their applicability for use in the AP1000. The safety analysis codes are LOFTRAN, LOFTTR2, NOTRUMP, WCOBRA/TRAC and WGOTHIC. The staff reviewed the report and concluded in a March 25, 2002, letter from J. E. Lyons (USNRC) to W. E. Cummins (Westinghouse), "Applicability of AP600 Standard Plant Design Analysis Codes, Test Program and Exemptions to the AP1000 Standard Plant Design," that the modified LOFTRAN and LOFTTR2 codes are acceptable for use in the AP1000 analysis with the following conditions and limitations:

- The Transients and accidents that Westinghouse proposes to analyze with the LOFTRAN code are listed in Table 2 of Enclosure of the March 25, 2002, letter and the NRC staff's review of LOFTRAN usage by Westinghouse was limited to this set. Use of the code for other analytical purposes will require additional justification.

- In the pre-application review, the NRC staff requested that Westinghouse perform main steamline break (MSLB) analyses for the AP1000 standard plant design. In particular, the staff wanted to assess the ability of the code to model the resulting steam formation in the reactor coolant loops. The applicant has provided this analysis. A review of this material is included in Chapter 21 of this report.

In addressing the staff review question regarding compliance with the limitations imposed by the staff on use of the LOFTRAN and LOFTTR2 codes, the applicant provides its response to the staff's RAI 440.054 and indicates that the codes are used only for those events identified in the NRC letter of March 25, 2002. The applicant has submitted the MSLB analysis in DCD Tier 2 Section 15.1.5. The analysis results demonstrate that voiding in the reactor coolant loops does not occur and, therefore, is not a concern for the MSLB event. Since the application of LOFTRAN and LOFTTR2 in the safety analysis for the AP1000 has complied with the limitations imposed by the NRC staff, the staff concludes that the application is acceptable.

The applicant also provides the assessment addressing the applicability of using NOTRUMP, WCOBRA/TRAC and WGOTHIC in the safety analysis for the AP1000. The staff has reviewed the applicant's compliance assessment and documented it's evaluation in Section 21.6.3 of this report for NOTRUMP, Section 21.6.4 for WCOBRA/TRAC, and Section 21.6.5 for WGOTHIC.

15.1.5 Steam Generator Mid-Deck Plate Induced Level Measurement Uncertainty

Westinghouse has issued three Nuclear Service Advisory Letter (NSAL), NSAL-02-3 and revision 1, NSAL-02-4 and 02-5, that document the problems with the Westinghouse-designed SG water level setpoint uncertainties. NSAL-02-3 and its revision, issued on February 15 and April 8, 2002, respectively, deal with the uncertainties caused by the mid-deck plate located between the upper and lower taps used for SG level measurements. These uncertainties affect the low-low level trip setpoint (used in the analysis for events such as the feedwater line break (FLB), ATWS and steam line break). NSAL-02-4, issued on February 19, 2002, deals with the uncertainties created because the void contents of the two-phase mixture above the mid-deck

plate are not reflected in the calculation and affect the high-high level trip setpoint. NSAL-02-5, issued on February 19, 2002, deals with the initial conditions assumed in the SG water level related safety analysis. The safety analysis may not be bounding because of velocity head effects or mid-deck plate pressure differential pressure that have resulted in significant increases in the control system uncertainties. In RAI 440.062 the staff requested the applicant to discuss: (1) how the AP1000 design accounts for all these uncertainties documented in these advisory letters in determining the SG water level setpoints, and (2) the effects of the water level uncertainties on the analyses of the LOCA and non-LOCA transients and the ATWS event.

The applicant response to RAI 440.062 stated that measurement uncertainties for the reactor protective system and engineered safety features actuation system instrument can be determined only when actual instrumentation is selected for the plant. The plant specific setpoint calculations will be completed and reviewed as part of the COL. The COL applicants referencing the AP1000 certified design will provide a calculation of setpoints for protective functions consistent with the methodology discussed in WCAP-14605, "Westinghouse Setpoint Methodology for Protective Systems, AP600." The methodology can be used for performing setpoint studies independent of the hardware used for the protection system, and therefore is applicable to the AP1000. The setpoint study will include applicable uncertainties discussed in the referenced NSAL. Using the methodology in WCAP-14605, plant nominal setpoints are calculated by adding the channel allowance from the setpoint study to the setpoint used in the safety analysis.

The COL applicant should evaluate and confirm the validity of the safety analysis documented in the DCD using plant specific setpoints and instrument uncertainties, including the SG mid-deck level measurement uncertainty. The COL applicants should submit in the plant specific applications the setpoint analysis and the associated safety analysis for the staff to review and approve. This is COL Action Item 15.1.5-1. The applicant should include this in the DCD. This is Open Item 15.1.5-1.

15.2 Transient and Accident Analysis

The applicant presents the results of transient and accident analysis for the AP1000 design in DCD Tier 2 Chapter 15. This section discusses the staff's evaluation of results of the analysis and the applicant's responses to the staff's RAIs. The staff's evaluation of the analysis for radiological releases is presented in Section 15.3 of this report.

15.2.1 Increase in Heat Removal from the Primary System

In DCD Tier 2 Section 15.1, the applicant presents the results of its analysis of the events involving an increase in heat removal from the primary system. The events include: (1) feedwater system malfunctions causing a reduction in feedwater temperature, (2) feedwater system malfunctions causing an increase in feedwater flow, (3) excess increase in secondary steam flow, (4) inadvertent opening of a SG relief or safety valve, (5) steam line break, and (6) inadvertent operation of the passive residual heat removal heat exchanger. The staff evaluates the analytical results and provides its evaluation in the following sections.

15.2.1.1 Decrease in Feedwater Temperature (DCD Tier 2 Section 15.1.1)

Decrease in feedwater temperature, a moderate-frequency event, may be caused by failure of a low- or high-pressure heater train. A reduction in feedwater temperature decreases reactor coolant temperature, that, in turn, causes an increase in core power because of the effects of the negative moderator coefficient of reactivity. Since the rate of energy change is reduced as load and feedwater flows decrease, the transient initiated from the zero-power conditions is less severe than the full-power case. The applicant's analysis for the limiting case is based on initial full-power conditions with a decrease of feedwater temperature caused by loss of one string of low-pressure feedwater heaters. The loss of a string of feedwater heaters results in a maximum reduction in feedwater temperature of 26.4 ^{o}C (79.5 ^{o}F.) The applicant's analysis indicates that the decrease in feedwater temperature results in an increase in core power of less than 10 percent of full-power. The decrease in feedwater temperature event is bounded by an excessive increase in secondary steam flow (a moderate-frequency event), that results in a power increase of 12 percent. The staff's review of the event with an excessive increase in secondary steam flow is discussed below in Section 15.2.1.3 of this report.

15.2.1.2 Increase in Feedwater Flow (DCD Tier 2 Section 15.1.2)

Increase in feedwater flow events may be caused by system malfunctions or operator actions that result in an inadvertent opening of a feedwater control valve. The excessive feedwater flow reduces reactor coolant temperature, which, in turn, causes a power increase because of the effects of the negative moderator coefficient of reactivity. Continuous addition of excessive feedwater is prevented by the steam generator high-2 water level signal trip, which closes the feedwater isolation valves and feedwater control valves and trips the turbine, main feedwater pumps, and reactor.

The applicant uses three codes to perform the analysis for this event. LOFTRAN is used to calculate the nuclear power transient, the RCS flow coastdown, and the primary pressure and temperature transient. FACTRAN is used to calculate the heat flux based on the nuclear power and flow from LOFTRAN and VIPRE-01 is used to calculate the departure from DNBR during the transient, using the heat flux from FACTRAN and the flow from LOFTRAN.

The applicant analyzes both the no-load case and full-power case. For the no-load condition, the applicant assumes a feedwater control valve malfunction results in a step increase to 120 percent of nominal feedwater flow to one SG. The feedwater temperature is assumed to be at a low value of 4.4 $^\circ$C (40 $^\circ$F). With the plant at no-load conditions, the turbine is not connected to the grid. Any subsequent reactor or turbine trip will not disrupt the grid and produce a consequential LOOP. Therefore, the applicant does not assume a LOOP in the no-load case. The results of the analysis shows that the no-load case is bounded by an uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition because of a lower reactivity insertion rate than the uncontrolled RCCA bank withdrawal event due to the effects of the negative moderator coefficient of reactivity. The analysis of an uncontrolled RCCA bank withdrawal event has been reviewed and approved by the staff in Section 15.2.4.1 of this report.

The applicant's analysis for the limiting case is based on initial full-power conditions with a increase of feedwater flow caused by malfunction of one feedwater control valve to its maximum capacity, resulting in a step increase to 120 percent of nominal feedwater flow to one SG. A reactor trip and an associated turbine trip are actuated on a SG high-2 level trip signal. In addressing the issue of a LOOP, the applicant assumes that a LOOP and the resulting coastdown of the RCPs occur 3 seconds after the turbine trip. As discussed in Section 15.1.3 of this report, the assumption of a LOOP with the delay time of 3 seconds is acceptable.

The applicant has considered plant systems and equipment discussed in DCD Tier 2 Section 15.0.8 that are available to mitigate the effects of the event. From the viewpoint of a SG overfilling, the worst case is a failure of the feedwater control valve in the affected SG to close in combination with a single failure of the feedwater isolation valve to close. In this case, the applicant indicates in its response to RAI 440.063 that a SG high-2 level trip signal will trip feedwater pumps and terminate the excessive feedwater flow. The staff notes that the feedwater pumps trip is a non-safety-related system. The staff has reviewed and approved the use of the feedwater pumps trip to terminate the excessive feedwater flow as discussed in Section 15.1.2 of this report.

The analysis was performed with the acceptable method and the results of the analysis demonstrate that the limiting full-power case meets the acceptance criteria for this moderate-frequency event. Specifically, the calculated peak RCS pressure is below 110 percent of the RCS design pressure, and the calculate DNBRs for the transient remain above the safety limit DNBR defined in DCD Tier 2 Section 4.4, thus, satisfying the acceptable criteria defined in Section 15.1.2 of the SRP. Therefore, the staff concludes that the analysis is acceptable.

15.2.1.3 Excessive Increase in Secondary Steam Flow (DCD Tier 2 Section 15.1.3)

Excessive increase in secondary steam flow may be caused by an operator action or an equipment malfunction in the steam dump control or turbine speed control. A rapid increase in steam flow results in a power mismatch between the reactor core power and the SG load demand.

The applicant analyzes four cases involving a 10-percent step load increase from rated load, using the previously approved LOFTRAN , FACTRAN, and VIPRE-01 codes, and assuming for:

Case 1 - minimum moderator feedback and manual reactor control,
Case 2 - maximum moderator feedback and manual reactor control,
Case 3 - minimum moderator feedback and automatic reactor control, and
Case 4 - maximum moderator feedback and automatic reactor control.

The 10-percent step load increase is the highest load increase allowed in the range of 25 to 100 percent full-power. Each case is analyzed without credit being taken for pressurizer heaters. At initial reactor power, the RCS pressure and temperature are assumed at their full-power values for the DNBR calculation. Uncertainties in initial conditions are included in the limit DNBR as described in WCAP-11397-P-A, "Revised Thermal Design Procedure." Steam flow

increases greater than 10 percent are analyzed in DCD Tier 2 Sections 15.1.4 and 15.1.5, and evaluated in Sections 15.2.1.4 and 15.2.15 of this report.

In demonstrating the capability of the plant for the cases with automatic rod control, the applicant takes no credit for deltaT trips on overpower and overtemperature. The applicant has considered plant systems and equipment discussed in DCD Tier 2 Section 15.0.8 that are available to mitigate the effects of the event, and determined that no single active failure in these systems or equipment will adversely affect the consequences of the event. In consideration of the effects of a LOOP, the applicant assumes a reactor trip with a coincident turbine trip followed by a LOOP 3 seconds later. The primary effect of the LOOP is to cause the reactor coolant pumps to coast down. Since the LOOP is delayed for 3 seconds after the turbine trip, the RCCAs are inserted well into the core before the RCS flow coastdown begins. The resulting power reduction compensates for the reduced flow encountered once power to the reactor coolant pumps is lost. Therefore, the applicant’s analysis indicates that the minimum DNBRs predicted during the event will occur prior to the time flow coastdown begins.

The results of the analysis show that the calculated peak RCS pressure is less than 110 percent of the design pressure, and the calculated minimum DNBR does not violate the safety DNBR limits. Since the analysis uses acceptable methods and the results meet the acceptance criteria of SRP Section 15.1.3 for this moderate-frequency event, the staff concludes that the analysis is acceptable.

15.2.1.4 Inadvertent Opening of a SG Relief or Safety Valve (DCD Tier 2 Section 15.1.4)

An inadvertent opening of a SG relief, safety, or steam dump valve may result in an increase in steam flow. The excessive cooldown increases, in the presence of a negative moderator temperature coefficient, positive reactivity which, in turn, increases the core power level.

In assessing the effects of the negative moderator temperature coefficient, the applicant’s analysis assumes the most negative moderator temperature coefficient corresponding to the end-of-life rodded core with the most reactive RCCA in its fully withdrawn position. Offsite power is assumed to be available to maximize the cooldown effect. Since the initial SG water inventory for the no-load case is greater, the magnitude and duration of the RCS cooldown resulting from steam releases is greater, and the associated positive reactivity addition is, therefore, also greater. Consequently, the applicant has determined that zero-power conditions are more limiting than at-power conditions for this postulated event. Since the turbine is initially in the trip condition for the plant at the zero-power, the consequential LOOP following the turbine trip is not considered a credible event and, therefore, is not modeled in the analysis.

The applicant has considered plant systems and equipment discussed in DCD Tier 2 Section 15.0.8 that are available to mitigate the effects of the event, and identified that the limiting single failure is a failure of one CMT discharge valve to open. The applicant also makes the following assumptions to maximize the cooldown effects:

- A typical capacity stem flow rate 236 kg/sec at 8.2 MPa (520 lbm/sec at 1200 psia) for any single steam dump, relief, or safety valve is assumed as the initial steam flow.
- The Moody model, without consideration of the piping friction losses, is used to calculate the steam flow.
- The most reactive RCCA is assumed to be stuck out of the core after the reactor trip.
- The lowest startup feedwater temperature is assumed.
- Four reactor coolant pumps are assumed to be initially operating.
- No moisture is assumed in the blowdown steam.
- Manual actuation of the PRHR system is assumed at the initiation of the event.

The applicant uses the LOFTRAN code to analyze the event. During the transient, the CMT injection and the associated tripping of the reactor coolant pumps are actuated automatically by the low cold-leg temperature "S" signal. Boron solution at 3400 ppm enters the reactor coolant system, providing negative reactivity to prevent a significant return to power and core damage. Later in the transient, as the reactor pressure continues to decrease, accumulators actuate and inject boron solution at 2600 ppm.

The results of the analysis show that the RCS pressure remains below 110 percent of the design pressure and departure from nucleate boiling (DNB) does not occur, thereby satisfying the acceptance criteria in Section 15.1.4 of the SRP. Therefore, the staff concludes that the analysis is acceptable.

15.2.1.5 Steam System Piping Failure (DCD Tier 2 Section 15.1.5)

A SLB, a limiting-fault event, is defined as a pipe break in the main steam system. The steam release during an SLB causes a decrease in the RCS temperature and SG pressure. In the presence of a negative moderator temperature coefficient, the RCS temperature decrease results in an addition of positive reactivity, that increases the core power level. The SG pressure decrease initiates a reactor trip when low pressure in the SG system produces a safeguards "S" signal. The "S" signal initiates the actuation of the CMTs, that, in turn, initiates a trip of the RCPs. In addition, the "S" signal isolates all feedwater control and isolation valves and trips the main feedwater pumps. The low cold-leg temperature signal isolates the startup feedwater control and isolation valves. The reactor is ultimately shutdown by the borated water from the CMTs.

The applicant uses the LOFTRAN code to calculate the system transient and utilizes the VIPRE-01 code to determine whether DNB has occurred for the core transient conditions calculated by the LOFTRAN code. A double-ended rupture at no-load conditions with no decay heat is analyzed as the limiting case. Because the SGs have integral flow restrictors with a 0.13 m^2 (1.4 ft^2) throat area, any rupture with a break greater than 0.13 m^2 (1.4 ft^2), regardless

of location, will have the same effect on the system as a 0.13 m^2 (1.4 ft^2) break, and so, this limiting break area is assumed in the analysis.

Because the average coolant temperature for a core tripped from at-power conditions is higher than at no-load and there is energy stored in the fuel, the RCS for a core tripped from at-power conditions contains more stored energy than at no-load. The additional stored energy reduces the cooldown caused by the SLB. Therefore, no-load conditions are more limiting than at-power conditions. To represent the limiting initial conditions and maximize the cooldown effect, the applicant assumes an initial condition for the SLB analysis of zero-power with no stored energy in the fuel.

The applicant has considered plant systems and equipment discussed in DCD Tier 2 Section 15.0.8 that are available to mitigate the effects of the event. For an SLB in which a single failure results in a failure of the MSIV in the intact SG to close, the applicant takes credit for closing the non-safety-related MSIV backup valves (including the turbine-isolation and control valves) to avoid an uncontrolled blowdown from two SGs. The use of the MSIV backup valves in the SLB analysis for backup protection is acceptable as discussed in Section 15.1.2 of this report. In addition, in order to maximize the overcooling effect, the applicant makes the following assumptions:

- The most reactive RCCA is in the fully withdrawn position after reactor trip.
- The end-of-life shutdown margin at zero-power is assumed when the accident is initiated.
- A negative moderator coefficient for the end-of-life rodded core with the most reactive RCCA stuck out is assumed.
- The Moody model without consideration of the piping friction losses, maximizing the blowdown flow rate, is used to calculate the steam flow.
- The maximum cold startup feedwater flow plus nominal 100-percent main feedwater is assumed.
- Four reactor coolant pumps are assumed to be initially operating.
- No moisture is assumed in the blowdown steam.
- Manual actuation of the PRHR system is assumed at the initiation of the event to maximize the cooldown.

Offsite power is assumed to be available to maximize the cooldown effect. The results of an SLB with offsite power available bounds the case with a LOOP for the following reasons:

- An initial condition of a LOOP results in an immediate RCP coastdown, which reduces the RCS cooldown effect and the magnitude of the return-to-power by reducing primary-to-secondary heat transfer.

- During the SLB event, the CMTs will be actuated to provide borated water that injects into the RCS. Flow from the CMTs increases if the RCPs have coasted down. Therefore, the analysis performed with offsite power and continued RCPs operation reduce the rate of boron injection into the core, that increases the potential of the return-to-criticality of the core after reactor trip.

- The plant protection system automatically provides a safety-related signal that initiates the coastdown of the RCPs coincident with CMT actuation. Since this RCP coastdown initiates early during the SLB event, the difference is insignificant in predicting the DNBRs for cases with and without offsite power.

- Because of the passive nature of the safety injection system, the LOOP will not delay the actuation of the safety injection system.

During the event, the reactor protection system initiates a trip of the RCPs in conjunction with actuation of the CMTs. The main steam isolation valves fully close in less than 10 seconds from receipt of a closure signal.

In response to the RAI 440.067, addressing the staff's concern regarding the effect of the timing of a LOOP on the analysis of the limiting SLB case, the applicant analyzes two full-break SLB cases initiated with the reactor at no-load conditions: (1) one with offsite power available throughout the event, and (2) one with offsite power loss simultaneous with the steam line break at the start of the event. The SLB analysis shows that for the case with LOOP, the RCPs begin coasting down at initiation of the transient, and for the case with offsite available, the protection system automatically trips the RCPs at 7.4 seconds into the transient. The results of the analysis show that the small difference in timing of the initiation of the RCP coastdown has no significant impact on the parameters that affected the return-to-power. The calculated peak core heat flux for the case with the offsite power available was slightly greater than that for the LOOP case (3.17 percent vs. 3.14 percent of the nominal full-power value). Consistent with the results presented in the DCD, this SLB analysis confirms that the SLB event initiated from the no-load conditions with offsite power available bounds the case with a LOOP initiated at time zero and is a limiting case.

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

- The applicant has used the LOFTRAN code for the system response determination and the VIPRE-01 code for the DNBR calculations in the analysis for an SLB event. Throughout the event, the RCS temperatures remains below saturated temperatures, confirming that the SLB analysis is within the applicable range of the LOFTRAN code (also see Section 15.1.4 of this report.)

- The values used for input parameters, resulting in a maximum cooldown effect and greatest potential of fuel failure, are conservative.

- The results of the SLB analysis have shown that the minimum DNBR remains above the allowable safety limit DNBR, and the peak RCS pressure remains below 110 percent of the design pressure, thus satisfying the acceptance criteria of SRP 15.1.5 for an SLB analysis.

15.2.1.6 Inadvertent Operation of the PRHR (DCD Section 15.1.6)

The inadvertent actuation of the PRHR system may be caused by an operator action or a false actuating signal that opens the valves that normally isolate the PRHR heat exchanger from the RCS. This moderate-frequency event causes an injection of relatively cold water into the RCS and results in an addition of positive reactivity in the presence of a negative moderator temperature coefficient.

The applicant considers plant initial conditions both at full-power and zero-power. A comparative assessment shows that the zero-power condition is bounded by the analysis performed for the inadvertent opening of a SG relief or safety valve event (discussed in Section 15.2.1.4 of this report). This is because the latter event, a moderate-frequency event, is analyzed assuming the PRHR heat exchanger actuation coincident with SG depressurization. Therefore, the applicant's analysis for the limiting case is based on initial full-power conditions.

The codes used by the applicant to perform this analysis are LOFTRAN for the system response calculation, FACTRAN for the heat flux determination, and VIPRE-01 for the DNBRs calculation. A negative moderator coefficient for the end-of-life rodded core is assumed. The core properties used in the LOFTRAN code for reactivity feedback calculations are generated by combining those in the sector with cold coolant nearest to the loop with the PRHR system with those associated with the remaining sector. Control systems are assumed to function for the condition for which their operation results in more severe conditions. Cases both with and without automatic rod control are considered. The reactor trips on high neutron flux, and overtemperature and overpower delta T trips are not credited in the analysis.

The applicant has considered plant systems and equipment discussed in DCD Tier 2 Section 15.0.8 that are available to mitigate the effects of the event, and determined that no single active failure in these systems or equipment will adversely affect the consequences of the event.

In considering of the effects of a LOOP, the applicant assumes that a reactor trip and an associated turbine trip occur at the time of peak power. A loss of power is assumed to occur 3 seconds after the turbine trip. Since the LOOP is delayed for 3 seconds after the turbine trip, the RCCAs are inserted well into the core before the reactor coolant system flow coastdown begins. The resulting power reduction compensates for the reduced flow encountered once power to the reactor coolant pumps is lost. The applicant's analysis indicates that the minimum DNBRs predicted during the event occur prior to the time flow coastdown begins. With the

assumption of no reactor trip occurrence during the transient, the results show that for the limiting case (the full-power case with manual rod control,) the core power stabilizes at about 108 percent of its nominal value.

The staff finds the assumptions used in the analysis are conservative for the reasons stated above, and therefore, acceptable. The results of the analysis for the limiting full-power case with and without offsite power available show that the RCS pressure remains below 110 percent of the design pressure, and the minimum DNBR remains above the safety limit DNBR, thus satisfying the acceptance criteria of the SRP for moderate-frequency events. Therefore, the staff concludes that the analysis is acceptable.

15.2.2 Decrease in Heat Removal by the Secondary System (DCD Tier 2 Section 15.2)

The applicant has analyzed transients specified in SRP Section 15.2 for cases resulting from a decrease in heat removal by the secondary system and identified the limiting cases with regard to the capability of the RCS boundary and fuel rod cladding to withstand the consequences of transients. The transients include: (1) steam pressure regulator malfunction or failure that results in decreasing steam flow, (2) loss of external electrical load, (3) turbine trip, (4) inadvertent closure of main stem isolation valves, (5) loss of condenser vacuum and other events resulting in turbine trip, (6) loss-of-ac power to the station auxiliaries, (7) loss of normal feedwater flow, and (8) feedwater system pipe break. The staff has reviewed the applicant's analyses and discussed the evaluation in Sections 15.2.2.1 through 15.2.2.8 below.

15.2.2.1 Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow (DCD Tier 2 Section 15.2.1)

There are no steam generator pressure regulators in the AP1000 design whose failure will cause a decreasing steam flow transient. Therefore, this event is not applicable to AP1000.

15.2.2.2 Loss of External Electrical Load (DCD Tier 2 Section 15.2.2)

The loss-of-external-electrical load, a moderate-frequency event, may be caused by electrical system failures. Following the loss of generator load, an immediate fast closure of the turbine control valves will occur. The transient in primary pressure, temperature, and water volume is caused by a decrease in heat transfer capacity from primary to secondary, due to a rapid decrease of steam flow to the turbine, accompanied by an automatic reduction of feedwater. The reactor is protected by the reactor trips on high pressurizer pressure, high pressurizer water level, and overtemperature delta T signals. The pressurizer and the SG safety valves may lift to protect the RCS from overpressurization.

This loss-of-external load event is bounded by the turbine trip event because the turbine control valves close more slowly than the turbine stop valves close as a result of a turbine trip event. The smaller reduction in heat removal due to a slower termination of steam flow will result in a lower peak RCS pressure. The staff's evaluation of the turbine trip analyses is discussed in Section 15.2.2.3 below.

15.2.2.3 Turbine Trip (DCD Tier 2 Section 15.2.3)

The turbine trip event may be initiated by signals resulting from a generator trip; low condenser vacuum; loss of lubricating oil; turbine thrust bearing failure; turbine overspeed; manual trip; and reactor trip. Following a turbine trip, the turbine stop valves rapidly close, and steam flow to the turbine abruptly stops. The loss of steam flow results in a rapid increase in secondary system pressure and temperature, as well as a reduction of the heat transfer rate in the SGs, which, in turn, causes the RCS pressure and temperature to rise.

The applicant performs the analysis for this event with the following codes: LOFTRAN for the transient response calculation, FACTRAN for the heat flux calculation, and VIPRE-01 for the DNBR calculation. In the DNBR determination, initial core power, reactor power and pressure, and RCS temperature are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions are included in the DNBR limit as described in WCAP-11397-P-A, "Revised Thermal Design Procedure." In maximizing the RCS overpressurization effects, the turbine is assumed to trip without actuating the rapid power reduction system. This assumption delays the reactor trip until conditions in the RCS result in a trip actuated by other signals. The reactor is assumed to trip by the first reactor trip setpoint reached on high pressurizer pressure, overtemperature delta T, high pressurizer water level, or low SG water level trip signals. In addition, no credit is taken for the turbine bypass system. Main feedwater is terminated at the time of turbine trip, with no credit taken for startup feedwater or the PRHR system to mitigate the consequences of the event. The pressurizer safety valves are assumed to be available to reduce the pressure increase during the transient. In consideration of the effects of a LOOP, the offsite power is assumed to last 3.0 seconds after the turbine trip. The applicant also has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and determined that no single active failure in these systems or equipment would adversely affect the consequences of the event.

In analyzing the turbine trip event, the applicant considers both minimum and maximum reactivity feedback cases. The applicant also considers the event with and without credit for the effect of pressurizer spray in reducing the reactor coolant pressure. Each case is analyzed with and without offsite power available. The results of the applicant's analysis show that the most limiting case analyzed is a turbine trip from full-power with minimum moderator feedback. The limiting case assumes no offsite power available and takes no credit for the effect of pressurizer spray in reducing the RCS pressure.

The staff finds that the NRC previously approved computer codes and adequate assumptions to maximize the peak pressure are used for the analysis, the calculated peak RCS pressure for the limiting turbine trip case is below 110 percent of the RCS design pressure, the pressurizer overfilling does not occur, and the calculated minimum DNBR is within the safety DNBR limit, thus satisfying the acceptance criteria of Section 15.2.3 of the SRP. Therefore, the staff concludes that the analysis is acceptable.

15.2.2.4 Inadvertent Closure of Main Steam Isolation Valves (DCD Tier 2 Section 15.2.4)

The inadvertent closure of steam isolation valves results in a turbine trip. The consequences of this event are the same as those of the turbine trip event discussed in Section 15.2.2.3 above.

15.2.2.5 Loss of Condenser Vacuum (DCD Tier 2 Section 15.2.5)

Loss of the condenser vacuum may result in a turbine trip and prevent steam from dumping to the condenser. Since the applicant assumes that the steam dump is unavailable in the turbine trip analysis, no additional adverse effects will result for the turbine trip event caused by loss of condenser vacuum. Therefore, the analytical results reviewed and discussed in Section 15.2.2.3 above for the turbine trip event also apply to the loss of condenser vacuum event.

15.2.2.6 Loss of AC Power to the Plant Auxiliaries (DCD Tier 2 Section 15.2.6)

The loss-of-ac power, a moderate-frequency event, may be caused by a complete loss of the offsite grid accompanied by a turbine-generator trip. From the decay heat removal point of view, this event is more severe than the turbine trip event because for this event, the decrease in heat removal by the secondary system is accompanied by an RCS flow coastdown, which further reduces the capacity of the primary coolant to remove heat from the core. The reactor will trip upon reaching one of the reactor trip setpoints in the primary and secondary systems as a result of the flow coastdown and decrease in secondary heat removal, or as a result of the loss of power to the control rod drive mechanisms.

The applicant uses the LOFTRAN code to perform the RCS system response analysis following a plant loss of offsite power. Only safety-related systems are credited in the analysis to mitigate the consequences of the event. In the system response analysis, the initial reactor power is assumed to be 102 percent of the rated power level. The ANSI 5.1 decay heat data are used to represent the core residual heat generation rate. A LOOP is assumed to occur at the time of the reactor trip, which is actuated on a trip signal of low steam generator (narrow range) level. The assumption of a LOOP coincident with the reactor trip is more conservative as compared to the case with the offsite power loss at time zero because of the lower SG water inventory for heat removal at the time of the reactor trip.

In addition, the PRHR heat exchanger heat transfer coefficients are assumed to be at low values associated with the low flow rate caused by the RCP trip. The staff requested in RAI 440.074 the applicant justify the adequacy of the calculated PRHR heat transfer coefficients during the loss-of-ac power event. The applicant's response to RAI 440.074 stated that the determination of the PRHR heat transfer coefficients was based on the same methods discussed in Chapter 9 of WCAP-12980, "AP600 Passive Heat Removal Heat Exchanger Test Final Report," that was previously reviewed and approved by the NRC for application to the AP1000 (NRC letter from J. E. Lyons to W. E. Cummins, dated March 25, 2002) as part of the AP1000 pre-application review. In WCAP-12980, the results of calculations using the Dittus-Boelter correlation show that the predicted values are in reasonable agreement with the PRHR test data. The operating conditions for the PRHR tests are: 1.1 Lpm to 37.8 Lpm (0.3 gpm to

10.0 gpm) for the single heat exchanger tube flow, 121 ^{0}C to 343 ^{0}C (250 ^{0}F to 650 ^{0}F) for the primary temperature and 0.446 MPa to 15.9 MPa (50 psig to 2300 psig) for the primary water pressure. The applicant calculates the PRHR tube flow and primary temperature during the loss-of-ac power event and shows that the flows and temperatures are within the range of the PRHR test conditions. During the loss-of-ac power transient, the primary pressure increased from 15.6 MPa to 17.3 MPa (2250 psia to 2500 psia,) which is slightly above test range.

Because the primary side fluid remains single phase during a loss-of-ac power event, the applicant indicates that the impact of pressure on the primary heat transfer coefficient is much less significant than that of the temperature. In addition, the applicant performs an analysis to address the effects of measurement uncertainties of the PRHR heat transfer coefficient on the plant behavior. In the analysis, the primary side heat transfer coefficient calculated using the Dittus-Boelter correlation is reduced by 25 percent. The results of the analysis show that reducing the PRHR primary side heat transfer coefficient by 25 percent results in a small reduction in the overall PRHR heat transfer rate, and that the reduction in heat transfer delays the time of the calculated peak values but does not significantly affect the magnitude of the calculated peak RCS pressure or peak pressurizer water volume. Since the applicant's analysis shows that the calculated PRHR tube flow and primary temperature are within the PRHR test range, and the calculated decay heat removal, the peak pressure and pressurizer water volume are not significantly affected by the magnitude of the PRHR primary side heat transfer coefficient during the low flow conditions of the loss-of-ac power events, the staff concludes that the PRHR heat transfer coefficient calculated at low flow conditions are adequate and acceptable for use in the loss of ac power event analysis.

The applicant uses the LOFTRAN, FACTRAN and VIPRE-01 codes with the revised thermal design procedure described in WCAP-11397-P-A, "Revised Thermal Design Procedure," dated April 1989, to perform DNBR calculations. In the analysis, initial reactor power, pressurizer pressure, and RCS temperature are assumed to be at their nominal values consistent with steady-state full-power operation. Uncertainties in initial conditions, as described in the revised thermal design procedure, are included in determining the DNBR limit during the transient. The SG safety valves and pressurizer safety valves are assumed to be functional for steam releases, and the CMTs are assumed to actuate when the PRHR HX cools down the RCS enough to set off a low cold-leg temperature "S" signal.

In considering the effects of a LOOP, the applicant assumes that the power loss and the resulting coastdown of the RCPs occurs 3 seconds after the turbine trip. If the LOOP occurs at the start of the event, the calculated DNBR transient will be the same as predicted for the event involving a complete loss of RCS flow, which is initiated by a LOOP at the beginning of the event. The results of the complete loss of RCS flow event are discussed in Section 15.2.3.2 below.

The applicant has considered plant systems and equipment discussed in DCD Tier 2 Section 15.0.8 that are available to mitigate the effects of the event, and determined that the worst single active failure is a failure to open one of the two valves in the PRHR discharge line. During the transient, the reactor trips on the low SG water level signal. AC power is assumed to be lost following the reactor trip. Loss-of-ac power causes the reactor coolant pumps to

coast down. The PRHR heat exchanger actuates on low narrow SG water level coincident with low startup feedwater flow rate, and the CMTs actuate when the PRHR HX cools down the RCS enough to set off a low cold-leg temperature "S" signal.

The results of the analysis show that the calculated minimum DNBR meets the safety DNBR limit, and the long-term PRHR heat removal capacity is sufficient to removal the decay heat. In addition, the results show that the peak RCS pressure does not exceed the RCS pressure limit, pressurizer overfilling does not occur, and the integrity of the RCS is maintained. Thus, the SRP acceptance criteria for the loss-of-ac power are met, and the staff concludes that the analysis is acceptable.

15.2.2.7 Loss of Normal Feedwater Flow (DCD Tier 2 Section 15.2.7)

A loss of normal feedwater flow, a moderate-frequency event, may be caused by feedwater pump failures, valve malfunctions, or loss-of-ac power sources. Following an event involving a loss of normal feedwater, the SG water inventory decreases as a consequence of continuous steam supply to the turbine. The mismatch between the steam flow to the turbine and the feedwater leads to the reactor trip on a low SG level signal. The PRHR HX will be actuated on either a low SG narrow range water level coincident with a low startup feedwater flow rate signal, or a low SG wide range water level signal. The PRHR HX transfers the decay heat to the in-containment refueling water storage tank (IRWST) and provides a continuous core heat removal capability following a loss of normal and startup feedwater. The RCS cooldown by the PRHR leads to actuation of a low cold-leg temperature “S” signal, which activates the CMTs. The CMTs inject the cold borated water into the RCS. Both the PRHR HX and CMTs provide heat removal capability for the long tem decay heat removal.

The applicant performs the analysis of this event with the NRC-approved LOFTRAN computer code. Initial reactor power is assumed to be 102 percent of the rated power. The relief of steam in the secondary system is assumed to be achieved through the SG safety valves. Upon initiation of the event, the RCPs are assumed to operate until they are automatically tripped by CMT actuation on a low cold-leg temperature "S" signal. In the analysis, only safety-related systems are assumed to function to mitigate the consequences of the events. The PRHR HX is actuated on a low SG wide range water level.

In consideration of the effects of measurement uncertainties, the initial temperature and pressurizer pressure are assumed at 4^{o}C and 0.446 MPa (7^{o}F and 50 psi) below the nominal values. In response to RAI 440.075 related to the staff’s concern about adequacy of the initial temperature and pressure assumed in the analysis, the applicant replied that during the loss of normal feedwater event, the ac power is assumed available after reactor trip and the CVCS makeup pump is assumed to operate. A lower initial pressurizer pressure results in a slightly higher CVCS flow rate that is calculated with the LOFTRAN code as a function of the RCS pressure that, in turn, results in a slightly higher peak pressurizer water level with a lower margin to pressurizer overfilling.   A lower initial RCS temperature results in a higher initial RCS mass and thus, a lower margin to the pressurizer overfilling. In addition, the applicant performed a sensitivity study and showed that the effects of the initial RCS temperature and pressurizer pressure on the plant transient behavior are insignificantly small. Therefore, the

staff concludes that the initial RCS temperature and pressure assumed in the analysis are acceptable.

The applicant has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and determined that the worst single active failure is a failure of one of the two valves in the PRHR discharge line to open.

In considering the effects of a LOOP, the applicant assumes that the power loss and the resulting coastdown of the RCPs occur 3 seconds after the turbine trip for DNBR calculations, The impact of the LOOP is to cause a coastdown of the RCPs. The applicant showed a loss of normal feedwater transient event followed by the consequential LOOP after turbine trip has the same scenario presented in DCD Tier 2 Section 15.2.6 for the loss-of-ac power analysis. Therefore, the calculated minimum DNBR, greater than the safety DNBR limits, for a loss-of-ac power event is bounding for a loss of normal feedwater event.

The analysis is performed with approved methods and the results of the analysis show that the calculated minimum DNBR meets the safety DNBR limit. For the long-term cooling, the analysis demonstrates that the PRHR can remove the core decay heat faster than it builds up during the transient and the long-term PRHR heat removal capacity is sufficient to removal the decay heat. In addition, the results show that the peak RCS pressure does not exceed the RCS pressure limit, the pressurizer overfilling does not occur, and the integrity of the RCS is maintained. Thus, the SRP acceptance criteria for the loss of normal feedwater event are met. Therefore, the staff concludes that the analysis is acceptable.

15.2.2.8 Feedwater System Pipe Break (DCD Tier 2 Section 15.2.8)

A FLB is defined as a break in a feedwater line large enough to prevent the addition of sufficient feedwater to maintain shell-side water inventory in the steam generators. The FLB may reduce the ability to remove heat generated by the core from the RCS because fluid in the SG is discharged through the break, and the break may be large enough to prevent the addition of main feedwater after the trip. A reactor trip may be actuated on signals of high pressurizer pressure, overtemperature delta T, low SG water level in either SG, low steamline pressure in either SG, and “High-2" containment pressure. During the event, the PRHR HX may be actuated on either a low SG narrow-range water level coincident with the a low startup feedwater rate signal or a low SG wide-range water level signal. The PRHR HX removes the decay heat and provides continuous core heat removal capability. The CMTs may be actuated on a low cold-leg temperature “S” signal.

The applicant assumes the break to occur in a feedwater line between the check valve and the SG with a double-ended rupture of the largest feedwater line. The double-ended break area assumed is 0.163 m^2 (1.755 ft^2). This break size is identified as the limiting break case because it results in the highest water inventory in the pressurizer and the highest peak primary pressure. The staff requested, in a follow-up to RAI 440.076, the applicant discuss the analysis used in determination of the limiting break case for the AP1000. In its response the applicant provided the results of a break size spectrum study for FLB events. The sensitivity study compares the results of the limiting FLB case presented in DCD with FLB analysis for break

sizes of 100 percent, 50 percent, 25 percent and 10 percent of the feedwater nozzle, performed assuming the break occurring at the initial transient time. The results confirm that the DCD case is the limiting case that results in the highest water inventory in the pressurizer and the highest peak primary pressure.

The analysis of this event is performed with the LOFTRAN computer code. The initial power is assumed at 102 percent of the rated power. Reactor trip is assumed to be initiated when the low SG narrow-range level point is reached on the affected SG. In minimizing the heat removal capability of the SG with the ruptured feedwater line, a saturated liquid discharge is assumed for the break fluid until all the water is discharged from the SG with the ruptured feedwater line. In minimizing the margin to the pressurizer overfilling, the initial pressurizer water level is assumed at a maximum allowable value. The applicant has considered the cases with a LOOP occurred simultaneously with the concurrence of the pipe break, the LOOP occurred during the FLB accident, and the cases without a LOOP, and identified (in the response to RAI 440.077) that the FLB with the LOOP occurred at the time of break is the limiting case, resulting in a highest RCS pressure. In the analysis for the limiting FLB case, the PRHR HX is assumed to actuate by the low SG water level (wide-range) with a maximum delay time of 15 seconds to initiate automatic alignment of the PRHR HX valves. In addition, the applicant takes no credit for the high pressurizer trip, for charging or letdown, or for energy deposited in RCS metal during the RCS heatup. During an FLB event, the engineered safety features (ESFs) required to function are the PRHR, CMTs, and steam isolation valves.

The applicant has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and determined that the worst single active failure is a failure of one of the two valves in the PRHR discharge line to open. In considering the effects of a LOOP on the DNBR calculations, the applicant assumes that the power loss and the resulting coastdown of the RCPs occur 3 seconds after the turbine trip.

The staff notes that the non-safety-related pressurizer spray is credited for heat removal to limit the increase in the peak RCS pressure. Also, a low pressurizer safety valve setpoint is assumed in the analysis. Both assumptions will result in a lower peak RCS pressure and thus, are non-conservative. The same non-conservative assumptions were made in the AP600 FLB analysis. During the previous AP600 review, the staff asked the applicant to reanalyze the FLB event and quantify the effects of the pressurizer spray and a low pressurizer safety valve setpoint on the results of the FLB event. In Westinghouse letter DCP/NRC 0962, dated July 18, 1997, the applicant replied that the event was reanalyzed without pressurizer spray operable and with the pressurizer safety valve setpoint at its normal value. The confirmatory analysis showed that the peak RCS was 18.08 mPa (2624 psia), an increase of 27.6 kPa (4 psi) as compared to the DCD case, and confirmed that the effects of the non-conservative assumptions on the calculated peak RCS pressure are small. Since the applicant's FLB analysis documented in the DCD shows that the RCS pressure response during a FLB event for the AP1000 design is similar to that of the AP600 design, the effects of the non-conservatism in modeling the pressurizer spray and pressurizer safety valves for AP1000 FLB analysis will be also small. In addition, the AP1000 FLB analysis shows that the calculated peak RCS pressure is less than 17.91 MPa (2600 psia). Therefore, the staff concludes that the calculated peak pressure demonstrates that the margin (greater than 1.03 MPa (150 psi)) to

the safety limit of 110 percent of the design pressure is sufficient to compensate for the non-conservative assumptions of the pressurizer spray and pressurizer safety valve models discussed above.

The FLB analysis is performed with the LOFTRAN computer code. The results of the analysis show that the peak pressures of the RCS and SG are below 110 percent of the design pressures, and the pressurizer does not overfill during the transient. For long-term cooling, the analysis demonstrates the core coolability by showing that the PRHR removes the core decay heat faster than it builds up. For DNBR calculations, a LOOP is assumed to occur 3 seconds after turbine trip and causes a coastdown of the RCPs. The applicant has shown that, for the first part of the transient up to the reactor trip and complete insertion of the control rods (where the minimum DNBR occurs), the system response for an FLB event is similar to the loss of ac power analysis presented in DCD Tier 2 Section 15.2.6. Therefore, the calculated minimum DNBR, greater than the safety DNBR limits, for a loss of ac power is bounding for the FLB event.

Since the applicant uses NRC-approved methods, the assumptions used in the analysis are adequate in maximizing RCS pressure and minimizing the calculated DNBRs, and the results show that the results of the analysis meet the acceptance criteria of the SRP15. 2.8 for the FLB break with respect to the pressure and safety DNBR limits, the staff concludes that the analysis is acceptable.

15.2.3 Decrease in Reactor Coolant System Flow Rate (DCD Tier 2 Section 15.3)

The applicant has analyzed the transients specified in SRP Section 15.3 for cases resulting from a decrease in RCS flow rate. The transients include: (1) partial loss of forced reactor coolant flow, (2) complete loss of forced reactor coolant flow, (3) reactor coolant pump shaft seizure (locked rotor), and (4) reactor coolant pump shaft break. The applicant has also identified the limiting case with regard to ability of the RCS boundary and fuel rod cladding to withstand the consequences of transients. The staff has reviewed the applicant’s analysis and discussed the evaluation in Sections 15.2.3.1 through 15.2.3.4 below.

15.2.3.1 Partial Loss of Forced Reactor Coolant Flow (DCD Tier 2 Section 15.3.1)

Partial loss of RCS flow, a moderate-frequency event, may be caused by a mechanical or electrical failure in a RCP, or by a fault in the power supply to the pumps supplied by an RCP bus. Protection against this event is provided by the low primary coolant flow reactor trip signal in any reactor coolant loop.

The applicant analyzes the partial loss of flow event with the following NRC-approved computer codes: LOFTRAN calculates the nuclear power transient, the primary system pressure and temperature transients, and the core flow during the transient based on the RCS loop coastdown flow from COAST; FACTRAN calculates the heat flux transient based on the nuclear power and flow from LOFTRAN; and VIPRE-01 calculates the DNBRs during the transient based on the heat flux from FACTRAN and the flow from LOFTRAN. The DNBR calculations are based on the revised thermal design procedure (RTDP) described in an NRC-approved

report (WCAP-11397-P-A). In the DNBR calculations, the initial reactor power, pressure, and reactor coolant system temperature are assumed to be at their nominal values, and the uncertainties in initial conditions are included in the DNBR limit as described in the RTDP. In maximizing the core power, and, thus, minimizing the DNBRs, the applicant uses the least negative moderator temperature coefficient and a large absolute value of the Doppler coefficient. The RCP flow coastdown is calculated based on RCS pressure losses and RCP characteristics. Reactor coolant fluid momentum is neglected to obtain a low coastdown flow, that will result in lower calculated DNBRs.

In consideration of the effects of a LOOP, the applicant assumes that the power loss and resulting coastdown of the RCPs occur 3 seconds after the turbine trip. In addition, turbine trip occurs 5 seconds following a reactor trip condition being reached. This delay on turbine trip is a feature of the AP1000 reactor trip system. The primary effect of the LOOP is to cause the remaining operating reactor coolant pumps to coast down. The analysis shows that the LOOP will have no effect on the calculated minimum DNBR since a rapid decrease in the heat flux following a reactor trip significantly compensates for the decrease in the RCS flow caused by a LOOP following a turbine trip, and the minimum DNBR occurs before initiation of a LOOP. The staff finds that the applicant's assumptions are conservative and thus acceptable.

The applicant has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and determined that no single active failure in these system or equipment adversely affects the consequences of the events.

Since an event involving the loss of three of the four RCPs is not credible, the consequences of the event were not analyzed. Also, the core flow would be much lower for an event involving a loss of two RCPs than for an event involving a loss of one RCP. Therefore, the results for an event with a loss of two RCPs are limiting and bound the event where only one RCP is lost. The loss of two RCPs and the loss of four RCPs are analyzed and discussed in DCD Tier 2 Sections 15.3.1 and 15.3.2.

The applicant analyzes the event with the NRC-approved methods and the results of the analysis for the limiting case, the loss of two RCPs, show that with and without offsite power available, the RCS pressure will remain within 110 percent of the design pressure, and the minimum DNBR will remain above the safety DNBR limit. The staff finds that the results of the analysis meet the acceptance criteria of SRP Section 15.3.1 regarding the limits for the calculated RCS pressure and the minimum DNBR. Therefore, the staff concludes that the analysis is acceptable.

15.2.3.2 Complete Loss of Forced Reactor Coolant Flow (DCD Tier 2 Section 15.3.2)

A complete loss of forced flow from RCPs may be caused by a simultaneous loss of electrical power to all RCPs. A LOOP and the resulting loss of all forced reactor coolant flow through the reactor core cause an increase in the average coolant temperature and a decrease in the margin to DNB. The signals of low RCP speed or the low reactor coolant loop flow will trip the reactor.

For the case analyzed with a complete loss of flow, the method of analysis and the assumptions made for initial conditions and reactivity coefficients are identical to those for a partial loss of flow, except that a reactor trip is actuated by the reactor coolant pump underspeed trip following the loss of power supply to all pumps at power. The methods and assumptions used in the analysis are discussed and found acceptable in Section 15.2.3.1 of this report. The results of the applicant's analysis show that the peak RCS pressure during the transient will remain below 110 percent of the system design pressure, and the calculated DNBR will remain above the design DNBR safety limit. Thus, the integrity of the RCS pressure boundary is not endangered, no fuel failure is predicted to occur, and core geometry and control rod insertability will be maintained with no loss of core cooling capability. Therefore, the staff determines that the analysis meets the acceptance criteria of SRP Section 15.3.2 with respect to the integrity of the RCS pressure boundary and the fuel rods, and concludes that the analysis is acceptable.

15.2.3.3 RCP Shaft Seizure (Locked Rotor) (DCD Tier 2 Section 15.3.3) and RCP Shaft Break (DCD Tier 2 Section 15.3.4)

RCP shaft seizure may be caused by an instantaneous seizure of an RCP rotor, and the RCP shaft break may be caused by an instantaneous failure of a RCP shaft. Both events are classified as limiting-fault events.

For both cases, the RCS flow through the affected reactor loop drops rapidly, leading to a reactor trip on a low-flow signal. After the reactor trip, energy stored in the fuel rods continues to be transferred to the coolant, causing the coolant temperature to increase and the coolant to expand. During this period, heat transfer to the shell-side of the SGs drops because the reduced flow results in a decreased SG tube film coefficient and the reactor coolant in the tube-side cools down while the shell-side temperature increases. The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the SGs, causes a pressure to increase throughout the RCS. The pressurizer safety valves will open to release steam from the pressurizer. The rapid decrease in the RCS flow also results in a decrease in the DNBR.

The analysis discussed in DCD Tier 2 Sections 15.3.3 and 15.3.4 indicates that the RCP shaft seizure event with a LOOP bounds the RCP shaft break event with a LOOP. This is because the RCP flow coastdown for the shaft seizure event is slightly faster, resulting in a lower minimum DNBR.

The more limiting RCP shaft seizure is analyzed for cases with and without offsite power available. For cases without power available, a LOOP is assumed to occur at 3 seconds following turbine trip. A LOOP causes a simultaneous loss of feedwater flow, condenser inoperability and coastdown of all RCPs. In the analysis, no credit is taken for restoration of offsite power before initiation of shutdown cooling.

The applicant has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and determined that no single active failure in these system or equipment adversely affects the consequences of the events. The analysis of this event is performed using the following NRC approved codes: the LOFTRAN code for the system response and the FACTRAN code for the heat flux calculation at the hot

spot. The reactor trip is actuated on the low reactor coolant flow signal. No credit is taken for the pressure-reducing effects of pressurizer spray, steam dump, or controlled feedwater flow.

The results of the analysis show that the maximum RCS pressure remains less than 110 percent of the design pressure. The applicant also indicates, in the response to RAI 440.080, that the calculated minimum DNBR is above the safety limit DNBR, and thus assures no rod failure. For the purpose of calculating dose releases, the applicant assumes that 16 percent of rods are damaged. The staff's evaluation of the radiological calculations is discussed in Section 15.3 of this report.

The applicant uses NRC-approved methods with results that show the peak RCS pressure will remain within 110 percent of the design pressure, and the radiological release will remain within the 10 CFR 50.34(a)(1)(ii)(D)(*1*) limits. Therefore, the staff finds that the analysis for the RCP shaft seizure event meets the acceptance criteria of SRP Section 15.3.3 and is acceptable.

15.2.4 Reactivity and Power Distribution Anomalies (DCD Tier 2 Section 15.4)

In DCD Tier 2 Section 15.4, the applicant presents the analytical results of events resulting from reactivity and power distribution anomalies. The transients include: (1) uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition, (2) uncontrolled RCCA bank withdrawal at power, (3) RCCA misalignment, (4) startup of an inactive RCP at an incorrect temperature, (5) a malfunction or failure of the flow controller in a boiling water recirculation loop that results in an increased reactor coolant flow rate, (6) chemical and volume control system malfunction that results in a decrease in the boron concentration in the reactor coolant, (7) inadvertent loading and operation of a fuel assembly in an improper position, and (8) spectrum of RCCA ejection accident. The applicant has also identified the limiting case with regard to ability of the RCS boundary and fuel rod cladding to withstand the consequences of transients. The staff's evaluation of the analytical results is as follows.

15.2.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-Power Startup Condition (DCD Tier 2 Section 15.4.1)

An uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition may be caused by a malfunction of the reactor control or rod control systems. Protection against this event is provided by the following reactor trips: the source range high neutron flux reactor trip, intermediate range high neutron flux reactor trip, power range high neutron flux reactor trips (low- and high-setting), and high neutron flux rate reactor trip.

For the analysis of this transient, the applicant uses TWINKLE for the average power generation calculation, FACTRAN for the hot rod heat transfer calculation, and VIPRE-01 for the DNBR calculation. The analysis assumes a conservatively small (in absolute magnitude) negative Doppler coefficient and the most positive moderator coefficient to maximize the peak heat flux. Reactor trip is assumed to occur on the low setting of the power range neutron flux channel at 35 percent of full power. A 10-percent uncertainty is added to the reactor trip setpoint value. The analysis assumes the maximum positive reactivity addition rate that is greater than that for the simultaneous withdrawal of the combination of two sequential RCCA

banks having the greatest combined worth at maximum speed of 1.14 m per minute (45 in. per minute). The most limiting axial and radial power shapes, associated with having the two highest-worth banks in their high-worth position, are assumed in the DNBR calculation. The initial power level is assumed to be below the power level expected for any shutdown condition (10^{-9} of nominal power.) The combination of the highest reactivity addition rate and lowest initial power produces the highest peak heat flux, resulting in a lowest calculated minimum DNBR, and is a conservative assumption.

The applicant has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and determined that no single active failure in these system or equipment adversely affects the consequences of the event. Since the turbine is initially in the tripped condition for the plant at a subcritical or low-power startup condition, a consequential LOOP following the turbine trip is not a credible event and, thus, is not modeled in the analysis.

The results of the analysis for this event show that the maximum heat flux is much less than the full-power value and that average fuel temperature increases to a value lower than the nominal full-power value. The calculated minimum DNBR is above the safety DNBR limits.

The staff has reviewed the assumptions related to the reactivity worths and reactivity coefficients used in the analysis and found that they maximize the heat flux and, thus, minimize the calculated DNBRs, and are conservative. The staff has reviewed the calculated consequences of this transient and found that they meet the requirements of GDC 10 in that the specified acceptable fuel design limits are not exceeded. The applicant also meets the requirements of GDC 20 in that the reactivity control system can be automatically initiated so that specified acceptable fuel design limits are not exceeded. In addition, GDC 25 is met in that a single malfunction in the reactivity control systems will not cause the specified acceptable fuel limits to be exceeded. Therefore, the staff concludes that the analysis satisfies the acceptance criteria of SRP Section 15.4.1, and is acceptable.

15.2.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (DCD Tier 2 Section 15.4.2)

An uncontrolled withdrawal of a RCCA bank in the power operating range, a moderate-frequency event, may be caused by a malfunction of the reactor control or rod control systems. The effect of such an event is an increase in fuel and coolant temperature (as a result of the core-turbine power mismatch). Plant protection is provided by reactor trips, including the high neutron flux trip, overpower and overtemperature delta-T trips, and pressurizer high pressure and pressurizer water level trips.

The analyses are performed with the following NRC-approved methods: the LOFTRAN code calculates the nuclear power transient, the flow coastdown, the primary system pressure transient, and the primary coolant temperature transient; the FACTRAN code calculates the heat flux based on the nuclear power and flow from LOFTRAN, and the VIPRE-01 code calculates the DNBR using the heat flux from FACTRAN and the flow, inlet core temperature and pressure from LOFTRAN. The DNBR calculations are based on the RTDP described in an

NRC-approved report (WCAP-11397-P-A). In the DNBR calculations, the initial reactor power, pressure, and reactor coolant system temperature are assumed to be at their nominal values, and the uncertainties in initial conditions are included in the DNBR limit as described in the RTDP. The maximum positive reactivity insertion rate is assumed to be greater than that for the simultaneous withdrawal of the combination of the two control banks, having the maximum combined worth at maximum speed. The high neutron flux signal is assumed to occur at 118 percent of nominal full-power. The overtemperature and overpower delta-T trips included instrumentation and setpoint uncertainties and the delays for trip actuation are assumed to be at the maximum values. The applicant analyzes cases with both minimum and maximum reactivity coefficients, and performs a sensitivity study of the effects of initial power levels (10-, 60-, and 100-percent power) and reactivity insertion rates (from 1 pcm/s to 110 pcm/s) on the consequences of the event.

The applicant has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and determined that no single active failure in these systems or equipment adversely affects the consequences of the event. In addressing the LOOP issue, the applicant assumes that the power loss and the resulting coastdown of the RCP flow occur 3 seconds after the turbine trip.

The results of the analysis show that the DNBR does not fall below the safety limit for all cases. Therefore, fuel integrity and adequate fuel cooling are maintained. The calculated peak RCS pressure will remain less than 110 percent of the design pressure. The staff finds that the analysis meets the acceptance criteria of SRP 15.4.2 with respect to the integrity of the fuel and pressure boundaries, and therefore, concludes that the analysis is acceptable.

15.2.4.3 Rod Cluster Control Assembly Misalignment (DCD Tier 2 Section 15.4.3)

RCCA misalignment incidents include one or more dropped RCCAs within the same group, a misaligned full-length assembly, and withdrawal of a single RCCA during operation at power. Misaligned rods can be detectable by asymmetric power distributions sensed by incore or excore neutron detectors or core exit thermocouples, by rod deviation alarms, or by rod position indicators. The deviation alarm alerts the operator to rod deviation from the group position in excess of 5 percent of span.

The applicant has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and determined that no single active failure in these systems or equipment adversely affects the consequences of the events. In consideration of the effects of a LOOP, the applicant assumes that a power loss and the resulting coastdown of the RCPs occur 3 seconds after the turbine trip.

The staff's evaluation of the analysis for a dropped full-length assembly, a misaligned full-length assembly, and withdrawal of a single RCCA during operation at power is as follows.

15.2.4.3.1 Analysis for a Dropped Full-Length Assembly

For an event with one or more RCCAs dropped from the same group, the core power decreases and the core radial peaking factor increases. The reduced core power and continued steam supply to the turbine cause the reactor coolant temperature to decrease. In the manual control mode, the positive reactivity feedback causes the reactor power to rise to the initial power level at a reduced inlet temperature with no power overshoot. In the automatic control mode, the plant control system detects the reduction in core power and initiates control bank withdrawal in order to restore the core power. As a result, power overshoot occurs, resulting in a lower calculated DNBR. The applicant determines that the automatic operating mode bounds the manual operating mode and is the limiting DNBR case.

The applicant analyzes the rod drop events in the automatic control mode using the LOFTRAN code for the system response and the VIPRE-01 code for the DNBR calculation. The results show that the calculated minimum DNBR is greater than the safety limit DNBR for any single or multiple RCCA drop from the same group and the peak RCS pressure will remain less than 110 percent of the design pressure. The staff finds that the analysis has satisfied the acceptance criteria of SRP Section 15.4.3 with respect to the minimum DNBR and peak pressure, and therefore, concludes that the analysis for the rod cluster control assembly drop event is acceptable.

15.2.4.3.2 Analysis for a Misaligned Full-Length Assembly

For RCCA misalignment situations, the applicant analyzes the two most limiting DNBR cases including: (1) RCCA misalignments in which one RCCA is fully inserted with the rest of the RCCAs at or above their insertion limits, and (2) a case in which a group is inserted to its insertion limit and a single RCCA in the group is stuck in the fully withdrawn position with the reactor at full-power conditions. In the DNBR analysis, the initial reactor power, pressurizer pressure, and RCS temperature are assumed to be at their nominal values consistent with steady-state full-power operation. The radial peaking factor associated with the misaligned RCCA for these two limiting cases is calculated by the applicant. Uncertainties in initial conditions as described in WCAP-11397-P-A, "Revised Thermal Design Procedure," are included in determining the DNBR limit during the transient. The analysis shows that the minimum DNBR is above the safety DNBR limit. Therefore, the staff concludes that the analysis is acceptable since it meets the acceptance criteria of SRP Section 15.4.3 with respect to the fuel cladding integrity.

15.2.4.3.3 Analysis for Withdrawal of a Single RCCA

The inadvertent withdrawal of a single assembly requires multiple failures in the rod control system, multiple operator errors, or deliberate operator actions combined with a single failure of the rod control system. Because of the low likelihood of the event to occur, the applicant classifies the single assembly withdrawal as an infrequent event for the AP1000 design. The event categorization is consistent with that approved by the staff for the Westinghouse operating plants, and therefore, is acceptable. The transient resulting from such an event is similar to that resulting from a bank withdrawal, but the increased peaking factor causes DNB to

occur in the region surrounding the withdrawn assembly. The radial peaking factor associated with the single RCCA withdrawal is calculated by the applicant using the VIPRE-01 code. Uncertainties in initial conditions as described in WCAP-11397-P-A, “Revised Thermal Design Procedure,” are included in determining the DNBR limit during the transient. In response to RAI 440.081, the applicant indicates that less than 4.0 percent of the rods in the core experience DNB during the limiting case, an event where RCCA rod banks are at the full-power rod insertion limits except one RCCA which is fully withdrawn. For the purpose of calculating dose releases, the applicant assumes that 5.0 percent of the fuel rods are failed. The assumption of fuel failure for the dose release calculation is more limiting than the guidance of SRP Section 4.4, which would only require failure of 4.0 percent of the fuel rods to be considered (i.e., the SRP states that all rods which experience DNB should be assumed to fail). Therefore, the staff concludes that the assumption is acceptable.

For the single rod withdrawal event (an infrequent event), the applicant has meets the requirements of GDC 27 by demonstrating that the resultant fuel damage is limited such that control rod insertability is maintained, and no loss of core coolability results. The DNBR calculation shows that a small fraction (4.0 percent) of the fuel rods may experience cladding perforation. Therefore, the staff concludes that the analysis is acceptable.

15.2.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (DCD Section 15.4.4)

Starting an idle RCP increases the injection of cold water into the core, that causes a reactivity insertion and subsequent power increase.

Because the TSs (DCD Tier 2 Chapter 16 TS 3.4.4) do not allow operation with a RCP inoperable for Modes 1 and 2, the applicant does not analyze this event at Modes 1 and 2.

15.2.4.5 A Malfunction or Failure of the Flow Controller in a Boiling-Water Reactor Loop that Results in an Increased Reactor Coolant Flow (DCD Tier 2 Section 15.4.5)

This section is not applicable to the AP1000 design.

15.2.4.6 Chemical and Volume Control System Malfunction That Results in the Boron Dilution in the Reactor Coolant (DCD Tier 2 Section 15.4.6)

The causes of an inadvertent boron dilution are failures of the demineralized water transfer and storage system (DWS) or CVCS because of control operator error or mechanical failure. The CVCS and DWS are designed to limit the dilution rate to values that allow sufficient time for automatic or operator actions to terminate the dilution before the shutdown margin is lost. The dilution rate is indicated by following indications: the indication of the boric acid and blended flow rates, the CVCS makeup pumps status, and the boron dilution rate deviation alarm. In Modes 1 and 2, either a rod insertion limit - low level alarm or an axial flux difference alarm will alert the operator to an unplanned boron dilution event. Furthermore, when the reactor is subcritical, the following alarms and indications are available to alert the operator to a boron dilution event: (1) high flux at shutdown alarm, (2) indicated source range neutron flux count

rates, (3) audible source range neutron flux rate, and (4) source range neutron flux multiplication alarm. Upon any reactor trip signal, source range flux multiplication signal, low battery charger input voltage signal or a safety injection signal, a safety-related function automatically isolates the potential unborated water from the DWS, and thereby terminates the dilution.

The applicant analyzed the boron dilution event for all modes of operation. The analysis was performed with the method consistent with that used in boron dilution event analysis for Westinghouse operating plants. The method consists of a generic fluid mixing model. The nodal scheme in the model includes a node to represent the RCS volume and a flow path to represent CVCS fluid transportation.

All cases discussed below assume that the dilution flow rate is 12.6 L/sec (200 gpm) of unborated water, which is the maximum makeup flow with both makeup pumps operating (as stated in DCD Tier 2 Section 9.3.6.6.1.2.)

15.2.4.6.1 Boron Dilution During Refueling (Mode 6)

Uncontrolled boron dilution is not a credible event during the refueling mode because administrative controls isolate the RCS from the potential source of unborated water by locking closed specified valves in the CVCS system during this mode of operation. Makeup water during refueling is supplied from the boric acid tank which contains borated water.

15.2.4.6.2 Boron Dilution During Modes 3, 4 and 5 of Operation

In Modes 3, 4 and 5, the analysis assumes a shutdown margin of 1.6 percent delta K/K (the minimum value required by the TSs for the shutdown modes) and the minimum initial reactor coolant volumes. Following the TS 3.4.8 requirements, the applicant assumes one RCP operating. In maximizing the effect of the boron dilution, the applicant uses the minimum amount of the water in the RCS to mix with the incoming unborated water. For Mode 3, the minimum RCS water volume assumed is the total RCS volume without the pressurizer and surge line, and the reactor vessel upper head volume. For Mode 4, the water volume assumed in the analysis is the water volume of the reactor vessel (RV) without the RV upper head volume when the normal residual heat removal system is used to remove the decay heat. For Mode 5, the water volume used in the analysis is the RCS water volume corresponding to the water level at mid-loop operations. The source range flux multiplication signal is assumed to actuate an alarm in the control room and closes the DWS isolation valves when the neutron flux increased by 60 percent over any 50-minute period (per item 15.a of TS Table 3.3.2-1.) The analysis shows that the automatic closure of the DWS isolation valves initiated by the source range flux multiplication signal occurs about 56.3 minutes after start of dilution for Mode 3, 12.3 minutes for Mode 4, and 12.03 minutes for Mode 5. The results of the analysis show that the automatic DWS valves isolation terminates the boron dilution and maintains the plant in a subcritical condition. The staff determined that the analysis meets the guidance in SRP 15.4.6 with respect to core criticality and concludes that it is acceptable.

15.2.4.6.3 Boron Dilution During Startup (Mode 2)

The plant is in the startup mode only for startup testing at the beginning of each cycle. During this mode of operation, the rod control is in manual control. The applicant performs an analysis of inadvertent deboration event at initial conditions representative of the startup mode of operation with an assumed unborated water flow rate of 12.6 L/sec (200 gpm). Following the TSs 3.1.1 and 3.4.4 requirements, the applicant assumes an available shutdown margin of 1.6 percent delta K/K and four reactor coolant pumps operating. The initial RCS water assumed in the analysis is the water volume includes all the RCS volumes except the pressurizer and surge line. Ten-percent tube plugging is accounted for in calculating the steam generator tube volume. The results of the analysis show that a reactor trip from a signal on the intermediate range neutron flux will: (1) initiate closure of the DWS isolation valves (DCD Tier 2 Table 15.4-1), (2) terminate the boron dilution, and (3) maintain the plant in a subcritical condition. Therefore, the staff determined that the analysis meets the guidance in SRP 15.4.6 with respect to core subcriticality and concludes that the analysis is acceptable.

15.2.4.6.4 Boron Dilution During Power Operation (Mode 1)

For Mode 1, the applicant analyzes both the manual mode and the automatic mode cases. For both cases, the initial RCS water volume used is the same as that for Mode 2 discussed above. For the manual mode case, the analytical result show that a reactor trip on the overtemperature delta T will initiate closure of the DWS isolation valves and terminate the boron dilution without a post-trip return-to-criticality occurring. Since a reactor trip isolates DWS valves and terminates the event, the subsequent LOOP assumption following a turbine trip (which occurs immediately after a reactor trip) as required by GDC 17 will not affect the results of the deboration event for the case in manual mode.

For the automatic mode case, an increase in the power and temperature caused by a boron dilution event is compensated by slow insertion of the control rods to avoid the reactor trip. Since a reactor and turbine trip does not occur as predicted in the analysis for the case in automatic mode, the subsequent LOOP event following a turbine trip (as required by GDC 17) is not a credible event, and thus, is not modeled in the analysis. For the AP1000 design, the redundant pre-trip alarms available to the operator for Mode 1 operation include a low-level rod insertion limit alarm and an axial flux difference alarm. The analysis shows that the available time interval from a low-low rod insertion limit alarm attributable to boron dilution to loss of shutdown margin is about 328 minutes (DCD Tier 2 Table 15.4-1.) The staff finds that the applicant has demonstrated compliance with the guidance of SRP Section 15.4.6 in that the redundant pre-trip alarms should alert the operator to initiation of the event in sufficient time (at least 15 minutes) to ensure detection of the boron dilution event during Mode 1 before possible loss of shutdown margin for a core with the control rods inserted.

The analysis shows that: (1) the inadvertent boron dilution events are prevented by the design and procedures during Mode 6, (2) for Mode 1 in a manual control mode and Modes 2 through 5, the automatic closure of the DWS isolation valves minimizes the approach to criticality and maintains the core in a subcritical condition, thus ensuring the integrity of the fuel and RCS pressure boundary, and (3) for Mode 1 in an automatic control mode, a number of alarms and

indications are available to alert an operator to a boron dilution event, and a sufficient time (328 minutes) is available for the operator to detect and terminate the event before loss of shutdown margin. Therefore, the staff determines that the analysis has satisfied the guidance in SRP Section 15.4.6 with respect to the operator action times and the core subcriticality, and therefore, concludes that it is acceptable.

In supporting the boron mixing model used in the analysis, the applicant specifies a required minimum core flow rate. Specifically, TS LCO 3.4.8 requires that at least one RCP be in operation with a total flow through the core of at least 630 L/sec (10,000 gpm) while in Modes 3, 4 and 5, whenever the reactor trip breakers are open. The staff requested in RAI 440.106 that the applicant provide the basis supporting a conclusion that the required core flow rate is sufficient to provide well-mixed flow condition assumed in the boron dilution analysis. The applicant replied that the process of selecting 630L/sec (10,000 gpm) includes general consideration of the results reported in NUREG/CR-2733, "Experimental Data Report for LOFT Boron Dilution Experiment L6-6," June 1982. As discussed in EGG-LOFT-5867, "Quick-Look report on LOFT Boron Dilution Experiment L6-6," May 1982, the key parameters of the loss-of-fluid test (LOFT) L6 series of tests were scaled based on the characteristics of the Westinghouse 4-loop Trojan PWR. With its four cold legs, the general configuration of the AP1000 inlet plenum region is similar to that of a 4-loop plant.

The LOFT test considered two low-pressure injection system flow rates that were scaled to provide equivalence to 189 L/sec (3000 gpm) and 378 L/sec (6000 gpm) residual heat removal (RHR) flow rates in the Trojan plant. Typical RHR related TSs that are intended to ensure adequate boron mixing in current Westinghouse designed plants, allow operation in the applicable mode with a single operating RHR pump. The results of the tests are documented in EGG-M-03783, DE83 013666, "PWR response to an Inadvertent Boron Dilution Event" (presented at the Third Multiphase Flow and Heat-Transfer Symposium Workshop, April 18-20, 1983.) The report indicates that for the 189 L/sec (3000 gpm) RHR flow equivalent case, "...the fluid volume in the reactor vessel was well mixed and that the assumption of perfect mixing, through not strictly correct, is adequate for calculational purpose." For the 378 L/sec (6000 gpm) flow equivalent case, the reported test results show an even closer approach to perfect mixing. These results of the LOFT tests have been used to support the typical plant TSs that generally accept an RHR flow in the vicinity of 189 L/sec (3,000 gpm) as being sufficient to justify the perfect mixing assumption modeled in the boron dilution analysis.

For AP1000, the minimum core flow required by the TS is much greater than the flow rates considered in the LOFT test and current accepted as providing adequate mixing in the operating plants. In addition, the surveillance requirement (SR) 3.4.8.1 places an operating speed requirement on a single RCP. Specifically, the SR requires that in order to be considered as an operating RCP, the single pump involved must be operating at a minimum speed of 25-percent rated speed, which produces a flow rate of 1239 L/sec (19,688 gpm.) This SR indicates that the total RCP flow is almost twice of the required 630 L/sec (10,000 gpm) core flow and much greater than 189 L/sec (3000 gpm) value that is typically applied to operating plants. Since the general configuration of the AP1000 inlet flow plenum region is similar to the that of a 4-loop Westinghouse plant, the LOFT tests that were scaled to a Westinghouse plant and used to support the boron mixing model for the current Westinghouse

plants are applicable to AP1000 for consideration of the selecting the minimum core flow rate to assure a well-mixed flow condition. Also, the required minimum RCP flow through the core of 630 L/sec (10,000 gpm) is much greater than the value of 189 L/sec (3000 gpm) that is typical applied to the operating plants and supported by the LOFT test results. Therefore, the staff concludes that the required minimum core flow provides reasonable assurance that it is sufficient to provide well-mixing flow conditions considered in boron dilution events, which were analyzed to address the guidance in SRP 15.4.6, and is therefore acceptable.

15.2.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position (DCD Tier 2 Section 15.4.7)

The applicant indicates that during fuel loadings, it will follow strict administrative controls to prevent operation with a misplaced fuel assembly or a misloaded burnable poison assembly. Nevertheless, the applicant performs an analysis of the consequences of a loading error.

The applicant uses the NRC-approved methods documented in WCAP-10965-P-A, “ANC: Westinghouse Advanced Nodal Computer Code,” dated September 1986, to perform the analysis for this event. In DCD Tier 2 Figures 15.4.7-1 through 15.4.7-4, the applicant provides comparisons of power distributions calculated for the nominal fuel loading pattern and those calculated for four loadings with misplaced fuel assemblies or burnable poison assemblies. The selected non-normal loadings represent the spectrum of potential inadvertent fuel misplacement including: (1) a case in which a Region 1 assembly is interchanged with a Region 3 assembly, (2) a case in which a Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly, (3) the enrichment error with a case in which a Region 2 fuel assembly is loaded in the core central position, and (4) a case in which a Region 2 fuel assembly instead of a Region 1 is loaded near the core periphery. Calculations include, in particular, the power in assemblies that contain provisions for monitoring with incore detectors.

The analysis described above shows that resulting power distribution effects will be either detected by the startup test involving the incore detector system (and hence remediable) or cause an acceptable small perturbation within the measurement uncertainty of 5 percent. The testing requirements and the results of the analysis demonstrate that the applicant has met the requirements of GDC 13 with respect to minimizing the possibility that a misloaded fuel assembly goes undetected (and minimizes the consequences of reactor operation in the event of inadvertent fuel misload). For the undetectable errors, the resulting power distribution changes are within the acceptable measurement uncertainty, ensuring no fuel failure and satisfying the SRP 15.4.7 guidance. Therefore, the staff concludes that the analysis is acceptable.

15.2.4.8 Spectrum of Rod Cluster Control Assembly Ejection Accidents (DCD Tier 2 Section 15.4.8)

The mechanical failure of a control rod mechanism pressure housing may result in the ejection of an RCCA. For assemblies initially inserted, the consequences are a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Although mechanical provisions have been made to render this accident extremely unlikely, the

applicant has provided its analysis of the consequences of such an event. The applicant has considered plant systems and equipment discussed in DCD Tier 2 Section 15.0.8 that are available to mitigate the effects of the event, and determined that no single active failure in these system or equipment adversely affects the consequences of the events. The staff has reviewed this analysis in accordance with SRP Section 15.4.8.

Methods used in the analysis are documented in WCAP-7588, Revision 1A, "An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods," dated January 1975, which the staff has previously reviewed and accepted.

The applicant analyzes two sets of cases for the rod ejection event: (1) one initiated at hot full-power (HFP) and (2) one initiated at hot zero-power (HZP). Both of these cases are analyzed using both beginning-of-cycle (BOC) and end-of-cycle (EOC) kinetics. The values of the initial plant parameters (power level, ejected rod worth delayed neutron fraction, and trip reactivity) assumed in the analysis are listed in DCD Tier 2 Table 15.4-3. The analysis credits the high neutron flux trip (high and low setting) to trip the reactor. The results show that the calculated hot spot radially averaged fuel enthalpy results for the four analyzed cases are: 181 calories per gram (cal/g) for HFP-BOC, 104 cal/g for HZP-BOC, 170 cal/g for HFP-EOC and 117 cal/g for HZP EOC. These values of peak fuel enthalpy are less than the safety limit of 280 cal/g specified in SRP 15.4.8, “Spectrum of Rod Ejection Accidents.” The calculated values are also within the Westinghouse-specified analysis limit of 200 cal/g. In addition, the calculated pressure surge resulting from the rod ejection does not exceed the RCS emergency limits (Service Level C), and, thus, satisfies the guidance of SRP 15.4.8 with respect to the RCS pressure limit.

In considering the effects of a LOOP, the applicant assumes that the power loss resulting in coastdown of the RCPs occurs 3 seconds after the turbine trip. The applicant has shown that the effect of a LOOP on the calculated minimum DNBR is negligible because a rapid decrease in the heat flux after the control insertion compensates for the decrease in the RCS flow caused by a LOOP, and the minimum DNBR occurs before initiation of a LOOP.

The analysis shows that less than 10 percent of the fuel rods experiences DNB as a result of the rod ejection event. For the purpose of calculating dose releases, the applicant assumes 10 percent of the fuel failure. The assumption of fuel failure for the dose calculation results in a higher radiological release dose and is conservative. Therefore, the assumption is acceptable.

Experimental data show failure of high burnup fuels at lower enthalpy values than the fuel enthalpy safety limit specified in SRP 15.4.8. However, there is broad agreement among the staff, the industry, and international community that burnup degradation in the margin to low-enthalpy fuel failure is likely to be regained by application of more detailed 3-dimensional (3-D) analysis methods of the fuel response to rod ejection accidents. Detailed 3-D models predict that the value of the peak fuel rod enthalpy would be below 100 cal/gm (“A Regulatory Assessment of Test Data for Reactivity-Initiated Accidents,” Nuclear Safety, Vol. 37, No. 4, October-December 1996, pp 271-288). In addition, a generic analysis performed by the applicant that assumes low-enthalpy fuel failure show that the radiological consequences of rod ejection accidents meet the acceptance criteria specified in SRP Section 15.4.8 (Appendix A).

As indicated in the response to RAI 440.181, the applicant generic analysis is predicated on conservative treatment of the experimental fuel data applied to existing and planned core operating within approved burnup limits for PWRs. Therefore, the staff concludes that, although the SRP 15.4.8 fuel enthalpy safety limit may not be conservative, the generic analysis provides reasonable assurance that radiological consequences of the rod ejection accident will not violate the acceptance criteria in SRP 15.4.8 for the AP1000 core operating within the current NRC approved burnup limits. The staff will not accept further extension of burnup limits until additional experimental information on fuel behavior is available to demonstrate that the fuel cladding will satisfy the regulatory acceptance criteria used in the rod ejection analyses for licensing applications. The review of the radiological releases is included in Section 15.3 of this report.

The analysis is performed with the acceptable methods. The results of the analysis show that the calculated values of peak fuel enthalpy are less than the acceptable limit specified in SRP 15.4.8, that the calculated peak RCS pressure does not exceed the RCS emergency limits (Service Level C), and that the radiological consequences meet the SRP 15.4.8 acceptance criteria. The staff finds that the analysis meets the acceptance criteria of SRP 15.4.8 with respect to the limits of the hot rod average enthalpy, RCS pressure and radiological consequences. Therefore, the staff concludes that the analysis is acceptable.

15.2.5 Increase in Reactor Coolant System Inventory (DCD Tier 2 Section 15.5)

In DCD Tier 2 Section 15.5, the applicant considers two cases which would result in an increase in the RCS inventory. These cases are (1) an inadvertent operation of the CMTs, and (2) malfunction of the chemical and control system.

15.2.5.1 Inadvertent Operation of the Core Makeup Tanks During Power Operation (DCD Tier 2 Section 15.5.1)

Spurious CMT operations at power can be caused by operator actions, a false electrical actuation signal, or a valve malfunction. The DCD presents the results of the most limiting case, a CMT inadvertently actuated by an operator error or a mechanical failure resulting in opening of two valves in the CMT discharge lines. During the event, the high-3 pressurizer water level signal occurs to trip the reactor followed by the PRHR actuation and eventually by an "S" signal, which then actuates the second CMT. The applicant analyzes the case using the LOFTRAN code. The following initial conditions are established by the applicant as a result of the sensitivity study to maximize the water level in the pressurizer:

- The reactor power is at 102 percent of nominal; the pressure is at 344.7 kPa (50 psi) below nominal and RCS temperature is at 3.9 °C (7 °F) below nominal.
- The pressurizer spray system and automatic rod control are operable.
- A least-negative moderator temperature coefficient, a low (absolute) Doppler power coefficient, and a maximum boron worth are assumed.

The CMT enthalpies are maximized to minimize the cooling provided by the CMTs. CMT injection and balance lines pressure drop is minimized to maximize the CMT flow injected into the primary system. Also, a minimum PRHR heat transfer is assumed for the decay heat removal. In response to RAI 440.085, the applicant indicates that the PRHR heat transfer capability has been minimized by modeling high pressure drops through the PRHR loop. A higher pressure drop limits the PRHR flow and reduces the calculated value of the primary side heat transfer coefficient. In addition, maximum TS value for PRHR tube plugging and minimum effective heat transfer area have been assumed. The assumptions using the higher CMT injection flow and a minimum PRHR heat transfer capability result in an increase in the RCS temperature and RCS expansion, thus reducing the margin to pressurizer overfilling. Therefore, the assumptions are conservative and acceptable.

The applicant has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and identified that one of the two PRHR parallel isolation valves failing closed is the worst single failure. In addressing the issue of a LOOP, the applicant assumes that a power loss and the resulting RCPs coastdown occur 3 seconds after the turbine trip.

The analysis assumes that the event is initiated by an inadvertent opening of the CMT discharge valves which results in the one CMT injecting borated water. During the transient, the reactor is tripped upon receipt of the “High-3" pressurizer level signal. Following reactor trip, the reactor power drops and average RCS temperature decreases with subsequent coolant shrinkage. At about the same time of the reactor trip, the turbine is tripped and after 3-second delay, a consequential LOOP is assumed and the RCPs is tripped. The cold-leg temperature increases and the CMT starts injecting cold water into the RCS at a much higher rate due to the increased driving head resulting from the density decreased in balance line. The CMT injection makes up the RCS shrinkage and within 1-minute after actuation of the “High-3" pressurizer level signal, the “High-3" pressurizer level setpoint is once again reached. The PRHR, with appropriate delay time, is then assumed to initiate. The primary and secondary pressures increase initially because of the assumed unavailability of the non-safety-related control systems, but eventually decreases as the PRHR removes the core decay heat. At about 1.39 hours the PRHR heat flux matches the core decay heat. During this period, the pressurizer level continues to slowly increase until the CMT recirculation decreases sufficiently to limit the mass addition to the RCS. After about 3.43 hours into the transient, the cold-leg temperature (“S”) setpoint is reached and the second CMT is actuated. The pressurizer level initially shrinks due to the addition of cold borated water. As the CMT continues to add water to the primary system, the pressurizer level begins to increase. At 3.69 hours, the first CMT stops recirculation. At 6.06 hours, the PHRH heat flux approaches the core heat flux. The CMTs stop recirculating at 8.52 hours into the transient.

The staff finds that the applicant uses the LOFTRAN code for the analysis, adequately identifies the limiting case and the results of analysis show that no RCS water is relieved through the pressurizer safety valves as a result of the transient. In addition, the calculated minimum DNBR remains above the safety limit value and the RCS and SG pressures remain below 110 percent of their respective design pressures. The staff determines that the analysis meets

the acceptance criteria of SRP Section 15.5.1 with respect to the pressure limit and core DNBR safety limit, and therefore, concludes that the analysis is acceptable.

15.2.5.2 CVCS Malfunction that Increases Reactor Coolant Inventory (DCD Tier 2 Section 15.5.2)

A CVCS malfunction may result in an event which increases RCS inventory. The CVCS malfunction may be caused by operator action, an electrical actuation signal, or valve failure. The DCD presents the results of the most limiting case, the CVCS malfunction caused by an operator error resulting in starting two CVCS pumps to deliver the flow to the RCS. The applicant has analyzed CVCS malfunction cases using the LOFTRAN code. The following initial conditions are established by the applicant as a result of the sensitivity study to maximize the water level in the pressurizer:

- The reactor power is at 102 percent of nominal; the pressure is 344.7 kPa (50 psi) above nominal and RCS temperature is at 3.6 °C (6.5 °F) above nominal.
- The pressurizer spray system is operable.
- A least-negative moderator temperature coefficient, a low (absolute) Doppler power coefficient, and a maximum boron worth are assumed.
- The initial boron concentration is chosen on the basis of an iterative analysis process, such that the limiting case bounds the cases that models explicit operator actions after the reactor trip.

The applicant has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and identified that the worst single failure is one of the two PRHR parallel isolation valves failing closed. In addressing the issue of a LOOP, the applicant assumes that a power loss and resulting RCPs coastdown occur 3 seconds after the turbine trip.

The analysis assumes that the event is initiated by a CVCS malfunction that results in injection from two CVCS pumps. As the CVCS injection flow increases RCS inventory, the pressurizer water volume begins increasing while the primary system is cooling down. The RCS temperature decreases to reach the low cold-leg temperature setpoint and actuates an "S" signal, resulting in a reactor trip. Following the reactor trip, the turbine is tripped and after 3-second delay, a consequential LOOP is assumed and the RCPs are tripped. Soon after the reactor trip, main feedwater lines, steamlines and the CVCS are isolated. After a delay of 12 seconds following the "S" signal, the CMT discharge valves are opened and 5 seconds afterward the PRHR HX is actuated. The operation of the PRHR HX and CMTs cools down the plant. At about 4.09 hours into the transient, the PRHR heat flux matches the core decay heat and at 5.61 hours, the CMTs stop recirculating.

The staff finds that the applicant uses the LOFTRAN code for the analysis with adequate inputs, appropriately identifies the limiting case, and the results show that no RCS water is

relieved from the pressurizer safety valves. In addition, the calculated minimum DNBR remains above the safety limit values, and the RCS and SG pressures remain below 110 percent of their respective design pressures. The staff determines that the analysis meets the acceptance criteria of SRP Section 15.5.2 with respect to the pressure lime and core DNBR safety limit. Therefore, the staff concludes that the analysis is acceptable.

15.2.6 Decrease in Reactor Coolant Inventory (DCD Tier 2 Section 15.6)

In DCD Tier 2 Section 15.6, the applicant provides analysis of events that may decrease the reactor coolant system inventory. These events are (1) an inadvertent opening of a pressurizer safety or inadvertent operation of the ADS, (2) a break in an instrument line or other lines from the reactor coolant boundary that penetrate the containment, (3) a steam generator tube failure and (4) a loss-of-coolant accident (LOCA) resulting from a spectrum of postulated piping breaks within the RCPB. The applicant's analysis and the staff's evaluation are discussed below.

15.2.6.1 Inadvertent Opening of a Pressurizer Safety or Inadvertent Operation of the automatic depressurization system (ADS) (DCD Tier 2 Section 15.6.1)

An accidental depressurization of the RCS may occur as a result of an inadvertent opening of a pressurizer safety valve or ADS valves. During the transient, the RCS pressure rapidly decreases and, in turn, causes a decrease in power because of the moderator density feedback. The pressurizer level may eventually drop far enough to cause a reactor trip on a low pressurizer level signal.

The ADS consists of four stages of depressurization valves which are interlocked such that stage 1 is initiated first with subsequent stages actuated only after previous stages have been actuated. The AP1000 design prohibits opening of the fourth-stage valves while the RCS is at nominal operating pressure. For inadvertent operation of the ADS valves, the applicant considers an opening of both first-stage ADS flow paths to be the limiting case because operation of these valves result in a greater depressurization rate than ADS stages 2 and 3 valves due to the shorter first-stage ADS valve opening time.

The applicant has also analyzed an inadvertent opening of the pressurizer safety valve. The flow area of the pressurizer valve is smaller than the combined two first-stage ADS valves; however, the safety valves open more rapidly than the ADS valves.

Normal reactor control systems are assumed not to function. The rod control system is assumed to be in automatic mode in order to maintain the core at full-power until the reactor trip protection function is reached.

The applicant has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and determined that no single active failure in these systems or equipment adversely affected the consequences of the event. In addressing a LOOP, the applicant assumes that a power loss and resulting RCPs coastdown occur 3 seconds after the turbine trip. The analysis shows that there is no effect of a LOOP on the calculated minimum DNBR since a rapid decrease in the heat flux after the reactor trip

compensates for the decrease in the RCS flow caused by the LOOP (which would follow a turbine trip) and that the minimum DNBR occurs before initiation of a LOOP.

The codes used by the applicant to perform the analysis for these events are LOFTRAN for the transient response calculation, FACTRAN for the heat flux calculation, and VIPRE-01 for the DNBR calculation. The DNBR calculations for these RCS valve opening events are performed using the revised thermal margin procedure in WCAP-11397-P-A. Initial core power, reactor power and pressure, and RCS temperature are assumed to be at their nominal values consistent with steady-state full-power operation.

The staff finds that the applicant analyzes the events using acceptable methods. The analysis shows that the overtemperature delta T reactor trip signal provides adequate protection against the RCS depressurization events. The calculated DNBR remains above the safety limiting value and the RCS pressure remains less than 110 percent of the design pressure throughout the transients. The staff determines that the analysis meets the acceptance criteria of SRP Section 15.6.1 with respect to the pressure and core safety limits, and therefore, concludes that it is acceptable.

15.2.6.2 Failure of Small Lines Carrying Primary Coolant Outside Containment (DCD Tier 2 Section 15.6.2)

The reactor coolant may be directly released from a break or leak outside containment in a CVCS discharge line or sample line. The applicant's analysis has identified that the worst case event is the double-ended break of the sample line between the isolation valve outside the containment and the sample panel. This sample line break results in the largest release of reactor coolant outside containment. The maximum break flow is limited to 8.2 L/sec (130 gpm) by the sample line orifices.

Both the isolation valves inside and outside containment are open only during sampling and the loss of sample flow will provide indication of the break to plant operators. A break in a sample line releases radioactivity and will actuate area and air radiation monitors. Since multiple indications are available for the operator actions, the applicant assumes that 30 minutes after initiation of a break, the operator would isolate the sample line and terminate further release of primary fluid discharged to atmosphere. The assumed operator action delay time of 30 minutes is consistent with the current operating plant design-basis analysis of a break of a small line outside containment, and therefore, is acceptable.

The assumptions used for analysis of this event are adequate and acceptable and the scenario described in DCD Tier 2 Section 15.6.2 ensures that the applicant has considered the most severe failure of piping carrying the primary coolant outside containment. In addition, the radiological releases are within the 10 CFR 50.34(a)(1)(ii)(D)(1) limits. Thus, the staff determines that the analysis meets the SRP 15.6.2 acceptance criteria. Therefore, the staff concludes that the analysis is acceptable. The staff's evaluation of the radiological release calculations is discussed in Section 15.3 of this report.

15.2.6.3 Steam Generator Tube Rupture (DCD Tier 2 Section 15.6.3)

The SGTR accident is defined as a penetration of the barrier between the RCS and the main steam system. This accident may be caused by the failure of a SG U-tube.

There are two parts of the analysis for the SGTR event: SG overfill calculation and the calculation of the SG mass releases used for the radiological consequence calculation.

The applicant has performed an analysis with LOFTTR2 to demonstrate that the AP1000 design features are capable of preventing the SG from overfilling with water. To maximize the SG water increase, the applicant identifies that the limiting single failure is the failure of the startup feedwater control valve to throttle flow when nominal SG level is reached. Other conservative assumptions maximizing the SG secondary water inventory are: high initial SG level, minimum initial RCS pressure, LOOP, maximum CVCS injection flow, maximum pressurizer heater addition, maximum startup feedwater flow and minimum startup feedwater delay time. The results of the analysis show that the effectiveness of the AP1000 protection system and passive system design features support the conclusion that an SGTR will not cause the SG to overfill with water.

For the SG mass release calculation, the applicant performs the SGTR analysis using the LOFTTR2 code for a case with complete severance of a single SG tube. At initiation of an SGTR, the reactor is assumed at nominal full-power. The initial secondary mass is assumed at nominal SG mass with an allowance for uncertainties. A LOOP is assumed at the start of the event because continued operation of the RCPs (resulting from a LOOP) has been determined to reduce flashing of primary-to-secondary break flow, consequently lower radiological releases. The reactor is assumed to trip due to the LOOP. Consistent with the assumption of a LOOP, main feedwater pump coastdown occurs after the reactor trip and no startup feedwater is assumed in order to minimize SG secondary inventory and, thus, maximize secondary activity concentration and steam release. The CVCS pumps are assumed to be loaded onto the diesel generators. Maximum CVCS flows and pressurizer heater addition are assumed at the initiation of the event (event though the offsite power is not available) to maximize primary-to-secondary leakage. The CVCS is assumed to isolate on the “High-2" SG narrow range level setpoint. Since the failure of the steam dump system would result in a steam release from the SG power-operated relief vales (PORVs) to the atmosphere following the reactor trip, the steam dump system is assumed to be inoperable to maximize the radiological releases.

The applicant has considered plant systems and equipment that are available to mitigate the effects of the event, as discussed in DCD Tier 2 Section 15.0.8, and identified that the most limiting single failure is a failed-open PORV on the affected SG. The applicant assumes that the single failure occurs coincidently with the low-2 pressurizer level signal, maximizing the integrated RCS-to-secondary break flow. The SG PORV is isolated when the associated block valve is automatically closed on a low steamline pressure protection system signal.

The analysis shows that after the reactor trip, a safeguard "S" signal is generated by low pressurizer pressure. The "S" signal results in CMT actuation and PRHR system actuation. Opening of the SG PORVs and operation of the PRHR and CMTs decrease the primary and

secondary pressures. When the secondary pressure decreases to the low steamline pressure setpoint, the steamline isolation valves and SG PORV block valves are closed. Following closure of the block valves, the primary and secondary pressures and faulted SG secondary water volume increase as break flow accumulates. This increase continues until the SG secondary level reaches the “High-2" narrow range level and isolates the CVCS pump. With continued RCS cooldown and depressurization provided by the PRHR system, primary pressure decreases to match the secondary pressure. At about 6.70 hours after the transient, the break flow terminates and the system reaches a stable condition. The analysis shows that the PRHR is capable of removing the core decay heat and preventing the unaffected PORV from opening. During the transient, the CMTs remain full and ADS actuation does not occur, and the SG does not overfill with water.

During an SGTR, the RCS depressurizes as a result of the primary-to-secondary leakage through the ruptured SG tube. The depressurization reduces the calculated DNBRs. The analysis shows that the depressurization before reactor trip for the SGTR is slower than for the RCS depressurization events discussed in Section 15.2.6.1 of this report. Following a reactor trip, the DNBR rapidly increases. Thus, the staff’s conclusion for the event discussed in Section 15.2.6.1 of this report also applies to the SGTR event in that the calculated DNBR remains above the safety limit.

For this analysis, the applicant uses the LOFTTR2 computer code together with conservative and acceptable assumptions to maximize the primary-to-secondary leakages. The results of the analysis show that the SG will not overfill with water, the maximum RCS will not exceed 110 percent of design pressure, and the minimum DNBR will remain greater than the safety DNBR limit. In addition, the analysis shows that long-term cooling can be achieved by the PRHR and CMTs, and the radiological releases will remain within the limits of 10 CFR 50.34(a)(1)(ii)(D)(*1*). The staff finds that the SGTR analysis meets the acceptance criteria of SRP Section 15.6.3 with respect to the pressure and core safety DNBR limits, and therefore, concludes that the analysis is acceptable. The staff's evaluation of the radiological release is discussed in Section 15.3 of this report.

15.2.6.4 Spectrum of Boiling Water Reactor Steam System Piping Failure outside Containment (DCD Tier 2 Section 15.6.4)

This section of the DCD is not applicable to the AP1000 design.

15.2.6.5 Loss-of-Coolant Accidents (DCD Tier 2 Section 15.6.5)

In DCD Section 15.6.5, Westinghouse presents the LOCA analysis results. The applicant’s analyses examine small break LOCAs, large break LOCAs, and post-LOCA long term cooling.

The applicant's LOCA analyses meet the following acceptance criteria for the calculated ECCS performance:

(1) The calculated peak cladding temperature (PCT) is less than 1204 °C (2200 °F).

(2) The calculated total oxidation of the cladding is within 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated is less than 0.01 times the hypothetical amount that can be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, are to react.

(4) Any calculated changes in core geometry will be such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended time required by the long-lived radioactivity remaining in the core.

These criteria are established to provide significant margin for ECCS performance following a LOCA. The staff finds that these acceptance criteria are consistent with the requirements of 10 CFR 50.46 (b)(1) - (b)(5) for ECCS performance and, therefore, are acceptable.

15.2.6.5.1 Small Breaks

The design of the AP1000 is that the reactor core can be kept cooled and covered with water by means of passive protective systems which do not require the start and operation of pumps to provide makeup water to the reactor systems. No operator action is required to actuate and control the passive protective systems. Active systems using pumps are also available and would be activated and controlled by the operator in the event of a SBLOCA. The operation of these active systems, however, is not credited in the design basis of the AP1000.

During a SBLOCA the AP1000 reactor system will depressurize to the pressurizer low-pressure setpoint, initiating a reactor trip signal. With further reduction in reactor system pressure the pressurizer low-pressure setpoint will be reached to actuate an “S” signal. The “S” signal causes the opening of valves in the discharge of the CMTs and PRHR. The CMTs will immediately begin to circulate borated water into the reactor vessel downcomer. Water will also begin to circulate through the PRHR heat exchanger to ensure decay heat removal. As the reactor system drains, the CMTs provide a source of water to replenish that lost out of the break. The “S” signal will also trip the RCPs. Tripping of the RCPs is important to retain water in the lower elevations of the reactor system, around the core and to minimize the loss of water from the break.

As the CMTs drain, signals are sent to the ADS valves to open in a prescribed sequence. The first three stages of ADS which are located at the top of the pressurizer will be sequenced first when the CMT water volume drops to 67.5 percent. The ADS-4 begins its open sequence when the CMT water volume drops to 20 percent.

The action of the break, PRHR, and ADS-1,2,3 causes the reactor pressure to decrease. When the pressure reaches 4.93 MPa (700 psig), the accumulator tanks which contain borated water pressurized with nitrogen will inject into the reactor vessel by way of the direct vessel injection (DVI) lines. Following actuation of the ADS-4 the reactor system pressure will approach that of the containment permitting borated water from the IRWST to flow by gravity into DVI lines and into the reactor vessel.

The applicant performed SBLOCA analyses using the NOTRUMP computer code. NOTRUMP calculates the flow of steam and water in one-dimension with variable nodalization. The code considers thermodynamic nonequilibrium between the steam and water phases. Significant code features include flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacking fluid nodes, and regime-dependent heat transfer correlations. The staff's evaluation and acceptance of the NOTRUMP code for AP1000 is discussed in Section 21.6.2 of this report.

The NOTRUMP analyses were made conservative by assuming decay heat at 120 percent of the ANS 1971 Standard as required by Appendix K to 10 CFR Part 50 of the Commission's regulations. The single failure of one of the four ADS-4 valves was assumed. This failure was determined to be the most limiting for the AP1000. The containment was assumed to remain at atmospheric pressure throughout the NOTRUMP analysis. The use of atmospheric pressure minimizes the mass flow out of the three ADS-4 valves that are assumed to remain operable which delays the time when reactor vessel reflooding from the IRWST can occur. To confirm the assumption that use of atomospheric containment pressure is conservative, Westinghouse performed an analysis using an assumed containment pressure of 274 kPa (25 psia). The use of the lower containment pressure was shown to be conservative. Other plant systems and equipment that are assumed available to mitigate the effects of the event, are discussed in DCD Tier 2 Section 15.0.8.

In the DCD, the applicant originally described SBLOCA analyses for the following cases:

(1) The inadvertent opening of both 4-inch ADS-1 valves

(2) A cold leg break of 2 inches equivalent diameter in the loop without the PRHR

(3) The double-ended rupture of a DVI line

(4) The double-ended rupture of a DVI line (274 kPa (25 psia) containment pressure)

(5) A cold leg break of 10 inches equivalent diameter

None of these breaks were calculated to cause core uncovery or core heatup. For the analysis of the 25.4 cm (10 in.) cold leg break, NOTRUMP calculated the core to become highly voided during the early part of the accident when the core stored energy was being removed. NOTRUMP does not have a detailed core heatup model for hot channel evaluation. To evaluate the core heating that might occur, the applicant performed a conservative heatup calculation in that that portion of the core that might experience critical heat flux was allowed to

heat adiabatically until the core void fraction was reduced by the combined flow from the two accumulators. This calculation resulted in a peak cladding temperature of 743 ^{O}C (1370 ^{O}F) which is much less than the 1204^{O}C (2200 ^{O}F) limit in 10 CFR 50.46 of the Commissions regulations. For break sizes larger than 25.4 cm (10 in.) equivalent diameter, even more core voiding and core heatup would be expected. These larger break sizes would be bounded by the large break sizes that are evaluated in Section 15.2.6.5.2 of this report.

In WCAP-14869, “MAAP/NOTRUMP Benchmarking to Support the use of MAAP4 for AP600 PRA Success Criteria Analyses,” dated April 1997, Westinghouse reported that for AP600 a small break in a hot leg would lead to a lower reactor vessel inventory than for a break in a cold leg of the same size. This is because for AP600, as is the case for AP1000, the cold legs enter the reactor vessel at a higher elevation than do the hot legs. The staff requested (RAI 440.098) that the applicant perform additional small break analyses including hot leg breaks for AP1000. In response the applicant provided the NOTRUMP predictions for the following small breaks.

A 5.08 cm (2 in.) cold leg break in the PRHR loop

A 5.08 cm (2 in.) hot leg break in the loop without the pressurizer

The double-ended break of a cold leg pressure balance line to a CMT

None of these break sizes resulted in core uncovery although the 5.08 cm (2 in.) hot leg break size resulted in a minimum inventory of water in the reactor vessel slightly less than the cold leg break of the same size.

Operating PWRs do not have ADS valves and PRHRs to depressurize and cool the reactor coolant system following a LOCA. Operating PWRs must cool and depressurize the reactor system using the SGs to remove decay heat. Under this scenario it has been postulated that water from steam condensation within the SGs tubes might flow into the lower cold leg elevations. The NRC staff does not believe deboration will occur during a SB LOCA for the AP1000 design. This issue is evaluated in Section 15.2.8 of this report.

As an additional check on the NOTRUMP results obtained by the applicant, the staff performed a series of audit calculations of SBLOCA for AP1000 using the RELAP5 computer code. RELAP5 is an advanced thermal/hydraulic simulation tool developed by the staff. Similar conservative assumptions were used in the RELAP5 analyses as were used by the applicant in the NOTRUMP analyses.   Decay heat was set at 120 percent of the ANS 1971 Standard. The single failure of one of the four ADS-4 valves was assumed. The containment was assumed to remain at atmospheric pressure. The core model in the RELAP5 analyses is somewhat more detailed than the NOTRUMP core model in that a hot rod is modeled with a higher heat flux than the average core. The increased heat flux of the hot rod allows for the possibility of fuel cladding heatup following DNB to be assessed.

The staff performed audit calculations for the following cases:

- The inadvertent opening of both 4-inch ADS-1 valves

- A cold leg break of 5.08 cm (2 in.) equivalent diameter in the loop without the PRHR
- A cold leg break of 8.89 cm (3.5 in.) equivalent diameter in the loop without the PRHR
- A hot leg break of 8.89 cm (3.5 in.) equivalent diameter in the loop with the pressurizer
- The double-ended rupture of a direct vessel injection line
- A cold leg break of 25.4 cm (10 in.) equivalent diameter

None of the breaks analyzed by the staff resulted in core uncovery or cladding heatup. RELAP5 calculated approximately the same minimum core water mass for all break sizes, however, slightly less core water mass was predicted for the double ended DVI line break than for the other breaks.

Based on the forgoing considerations, the staff concludes that the applicant’s analyses for a spectrum of small piping breaks in the reactor pressure boundary is acceptable and meets the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50, and that the calculated performance of the passive emergency core cooling system following a postulated SBLOCA is acceptable.

15.2.6.5.2 Large Breaks (DCD Tier 2 Section 15.6.5.4A.8)

The applicant performed the LBLOCA analyses using the WCOBRA/TRAC code as documented in WCAP-12954, "Code Qualification Document for Best Estimate LOCA Analysis." WCOBRA/TRAC is the Westinghouse's “best estimate” (BE) thermal-hydraulic computer code used to calculate thermal-hydraulic conditions in the reactor system during blowdown and reflood of a postulated LBLOCA. This code is comprised of the BE features needed to satisfy the requirements of 10 CFR 50.46(a)(1)(i) for a realistic code.

In addition, WCOBRA/TRAC is used to analyze the post-LOCA long term cooling of the AP1000 using 10 CFR Part 50, Appendix K, decay heat assumptions. The staff's evaluation and conclusions for the LTC is discussed Section 15.2.7 of this report.

The applicant used the WCOBRA/TRAC code to perform the LBLOCA analysis. DCD Tier 2 Table 15.6.5-4 lists the initial plant physical configuration, power-related parameters, initial fluid conditions, and RCS boundary conditions used to determine the most limiting break size. These initial conditions are determined from the applicant's sensitivity study of the worst-case set of combinations that result in a highest limiting calculated PCT. To determine the limiting break case, the applicant performed parametric studies for the PCT with respect to bounding initial conditions and associated uncertainties using the methods described in WCAP-14171, Revision 2, “WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident,” dated March 1998, to calculate the 95th percentile PCT. The results of the analysis show that the DECLG break results in a maximum PCT and is the limiting case. In all cases analyzed, the bounding core design values of Fq = 2.60 and FdH = 1.65 are applied to the hot rod, at 102

percent of nominal core power. Finally, it was noted that, in the search for the limiting large break LOCA, the hot leg break and the cold leg limiting-split break were included.

The applicant considered plant systems and equipment that are available to mitigate the effects of the accident, as discussed in its response to RAI 440.097, Revision 1, and identified the limiting single failure as a failure of one CMT discharge valve to open. In modeling the CMTs and accumulators, the applicant minimized the capability to add borated water by assuming the failure of one CMT discharge valve to reflect the limiting single failure.

The applicant presents the results of the LBLOCA analyses in the DCD Tier 2 Tables 15.6.5-5 through 15.6.5-8 and Figures 15.6.5A-1 through 15.6.5A-12. Additional information was submitted in RAI responses 440.097, Revision 1, Table 15.6.5-8 and Figures 440.097R1-1 through 440.097R1-3. Following a LBLOCA, the reactor trip actuates on low pressurizer pressure trip signal. The insertion of the control rods is not credited in the LBLOCA analysis. Within a few seconds after the initiation of a LBLOCA, an "S" signal actuates on the containment “High-2" pressure. As a result, after appropriate delays, the PRHR and CMT isolation valves open and containment isolation occurs. The rapid depressurization of the RCS during a LBLOCA leads to the initiation of accumulator injection early in the transient. The accumulator flow reduces CMT delivery to the degree that the CMT level does not reach the ADS stage-1 valve actuation setpoint until after the accumulator tank empties following completion of the blowdown phase. The applicant's calculations continue until the fuel rods are quenched.

The applicant uses the WCOBRA/TRAC models to perform the LBLOCA analyses with calculated PCT uncertainties that are derived from the effects of model-related parameters while the initial condition-related parameters used in analyses are bounding and conservative values for the AP1000.

The applicant addressed the limitations in WCOBRA/TRAC relating to the PCT for values greater than 940 ^{O}C (1725 $^{\circ}$F). Staff review of the sensitivity calculations required by the code-limitation indicated that the results reinforce the conservatism of the calculation.

15.2.6.5.2.1 Summary of the Large Break LOCA Analysis Results

As per 10 CFR 50.46 the AP1000 large break LOCA analysis showed that there is a high level probability that the following criteria are met:

- The calculated PCT will not exceed 1204 ^{O}C (2200 $^{\circ}$F).
- The calculated maximum cladding oxidation will not exceed 0.17 percent the total cladding thickness before oxidation.
- The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam will not exceed one percent of the amount which would be generated if the entire cladding metal surrounding the fuel (excluding the cladding surrounding the plenum volume) was to be oxidized.

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

Therefore, the calculated results of the AP1000 large break LOCA satisfies the requirements of 10 CFR 50.46, and therefore, are acceptable.

15.2.7 Post-LOCA Long-Term Cooling

The purpose of the LTC analysis is to establish that the passive cooling mode (absence of active components or beneficial operator intervention) provides adequate core cooling until the plant is recovered. In this context, plant recovery means that the reactor is in a safe and stable configuration under operator control. In addition it must be established that sufficient water flows in and out of the vessel to cool the core and prevent boron precipitation during the LTC phase. Boron precipitation could obstruct water passage from the downcomer to the core. The regulations have no provisions for the LTC in a passive cooling mode. The applicant proposed to apply 10 CFR Part 50 Appendix K methodology to the extent possible because the LTC is a benign slowly evolving transient. In this context the applicant applied the ANS 1971 + 20 percent decay heat source. This is conservative and is acceptable.

The applicant used WCOBRA/TRAC for the LTC analysis. WCOBRA/TRAC was qualified (using the Oregon State University (OSU) experimental results, see WCAP-14776, Revision 4, “WCOBRA/TRAC OSU Long-Term Cooling Final Validation Report,” April 1998) for AP600 LTC calculations. In response to staff RAI 440.091, Revision 1, the applicant presented data and information which demonstrated that WCOBRA/TRAC is capable of addressing the parameter values in AP1000. The parameter of primary interest here is the flow regime through ADS-4 and its ability to entrain liquid droplets during LTC. The applicant presented calculations to show that steam velocity through ADS-4 in AP600, AP1000 and the OSU experiments are all in the same flow regime. The OSU results demonstrated liquid entrainment. In addition, the applicant mapped these points on a flow-regime map for a vertical pipe to demonstrate that all of the above data are similar and reinforce the conclusion that droplet entrainment will take place. (Wallis, G. B., “Phenomena of Liquid Transfer in Two-Phase Annular Flow,” International Journal of Heat and Mass Transfer, Volume 11, pp. 783-785, 1968.)

In AP1000 the LTC phase initiates with the stabilization of the IRWST injection which is considerably latter (1700 to 1800 sec) than the end (about 230 sec) of the reflood phase of LOCA in conventional plants. At the staff’s request the applicant addressed this in-between time in the response to RAI 440.091, Revision 1. The staff did not find that any special qualification requirement was needed to apply WCOBRA/TRAC in this phase of the transient. WCOBRA/TRAC is qualified for the LOCA and the LTC phases (i.e. before and after) and there are no new or different phenomena in the in-between phase of the transient to require special code qualification.

In the DCD Tier 2 Section 15.6.5.4C.2, "DEDVI Line Break with ADS Stage Four Single Failure, Passive Core Cooling System Only Case; Continuous Case," and the response to RAI 440.091, Revision 1, the applicant described: the core inlet and outlet liquid mass flow rates and downcomer and core collapsed liquid levels. The PCT during LTC is around 145 ^{O}C (300 °F) and the core collapsed liquid level is in the range of 1.52 m to 3.05 m (5 to 10 ft) above the bottom of the core. The staff finds these conditions acceptable in that the fuel is at a relative low temperature and a large part of the core is covered with water.

As indicated above the stabilization of IRWST injection flow is considered the beginning of the LTC phase. IRWST injection follows the automatic depressurization system blowdown (ADS stages 1-4). In part the blowdown is achieved by water or steam leakage through the break. IRWST injection provides the initial post-LOCA sustained LTC of the core as cooling water enters the RCS via the DVI lines from the IRWST and exits the RCS via the break and the ADS valves. When the IRWST drains to the "low-3" level, the sump isolation valves open initiating vessel sump injection. Water will be boiling in the vessel at this time and the sump water temperature will be rising to near the saturation temperature. Steam transports core heat to the containment shell and steam condensation returns the water to the IRWST or the sump. Heat rejection from the containment shell to the environment provides the ultimate heat sink. The cycle so established continues until the plant is recovered under operator control.

The accumulators, the CMTs and the IRWST all contain borated water. All of their contents are eventually spilled into the sump which is the final source of cooling water during the LTC phase. During this quasi-steady state LTC phase, cooling water enters the vessel and part of it exits as steam and part as liquid. (The LTC is a quasi-steady state because decay heat, the driving force, continues to decrease). The boron concentration in the vessel will increase to an equilibrium value depending on the steam/water mass ratio exiting the vessel. If all of the cooling water is evaporated in the vessel, the boron concentration would continuously increase. The equilibrium concentration is inversely proportional to the mass-fraction of liquid exiting the vessel. Depending on the temperature of the water in the vessel and the corresponding boric acid solubility, the equilibrium concentration could increase above the original sump concentration but not to the values which could result in boron precipitation.

15.2.7.1 The Window Method

WCOBRA/TRAC is a complex code requiring large amounts of computation time to track a transient. The LTC phase (regardless of the break that initiates the transient in the AP1000) is a slowly evolving, extremely long transient not experienced in any existing type of reactor. For these reasons, the applicant has used a "window" method for analysis of the transients. The analysis of the transient is accomplished using time-segments called "windows" between 5,000 to 10,000 seconds with the start and stop times chosen to encompass the most important portions of the transient from a safety perspective. The basis for the validity of the window method is that, all other factors being the same, the thermal-hydraulic phenomena are driven by the level of the decay heat. This was shown with extensive calculations for the AP600.

15.2.7.2 LTC Analyses and Results

Through analyses performed for the AP600, it was established that the evolution of the LTC is independent of the initiating transient and the determining parameters are the decay heat, cooling water flow and steam-water flow resistance. At the initiation of the LTC, the core has been quenched, the accumulators and the CMTs have emptied, and the IRWST injection has stabilized. At this stage the objective is to demonstrate that the passive system is capable of removing the decay heat. Therefore, the limiting case has the highest decay heat, the lowest cooling water flow, and the highest resistance to steam-water mixture exiting the vessel. As in the AP600 the parameters of the limiting case occur in the DEDVI line break. DCD Tier 2 Section 15.6.5.4C.2 presents a DEDVI line break with an ADS-4 single failure. Initial containment pressure was derived from the WGOTHIC code, and the transient was carried to 9,000 seconds until after a quasi-steady state sump recirculation was established. The results are presented in DCD Tier 2 Figures 15.6.5.4C-1 through 15.6.5.4C-28. The results show that the fuel PCT is low, and the water circulation is adequate to provide core cooling.

With regard to the boron precipitation issue, the results presented in DCD transient analysis (1) did not quantify the amount of water exiting the vessel; (2) there was no clear indication of void distribution in the core; (3) did not characterize the water-steam mixture flow regime in the ADS-4; and (4) did not minimize the steam velocity through the ADS-4. At the staff's request, the applicant presented a more conservative case by assuming that all ADS-4 valves are open and the containment pressure is at a maximum. In addition, the applicant presented a qualification of the WCOBRA/TRAC model regarding ADS-4 water-steam flow (RAI responses to 440.091, Revision 1). The staff reviewed this information and (as stated above) found that there is adequate justification for the WCOBRA/TRAC ADS-4 flow model. The applicant demonstrated that the flow regime is the same as in AP600 (annular flow) which would entrain fluid particles to expel water from the vessel as required to avoid boron concentration in the vessel and/or precipitation. The amount of water to be removed from the core was quantified. In addition, literature was cited regarding flow regimes applicable to the conditions of the ADS-4 which reinforced the credibility of the results.

However, the applicant did not present a detailed enough case regarding void distribution in the core. Persistent voiding in the core could result into adiabatic heating of the fuel. This is Open Item 15.2.7-1.

15.2.7.3 DEDVI Line Break and Wall-to-Wall Flooding

DCD Tier 2 Section 15.6.5.4C.3 describes a limiting case of the DEDVI line break transient regarding the amount of water available in the sump to maintain passive cooling. It is assumed that the dry spaces below flood level are flooded, thus, lowering the water level in the case of sump recirculation. The applicant estimated that 28.5 days will be required for flooding to take place. The window method was applied for the calculation of the transient at that time, assuming the water level accounted for flooding of the dry spaces and decay heat at 28.5 days. The results showed that the core remained cooled and the steam velocity was able to sustain water droplet entrainment.

15.2.7.4 LTC Summary

The staff has reviewed DCD Tier 2 Section 15.6.5.4C, “Post-LOCA Long-Term Cooling,” with respect to both core coolability and potential for boron precipitation. The physical phenomena which appear during AP1000 LTC transients are not normally encountered in the early part of the LOCA transient: blowdown, refill, quench and reflood. The only significant characteristic retained in this analysis from Appendix K to 10 CFR Part 50 is the decay heat source strength. The review indicated that the applicant chose conservative conditions for the analysis. The results demonstrate that, in all instances, there is sufficient liquid and steam coolant flow to ensure core cooling and provide core flushing to avoid boron concentration in the vessel.

However, the information provided regarding core void distribution was not sufficiently detailed to draw the conclusion that adiabatic heating would be avoided. The applicant should provide more nodes in the vessel axial void distribution analysis during LTC to demonstrate that there is no possibility of adiabatic fuel heating. This is Open Item 15.2.7-1.

15.2.8 Deboration during SBLOCAs

The staff reviewed the issue of boron dilution associated with the SBLOCA reflux condensation, the so called “Finnish Scenario.” In response to a staff request for additional information, RAI 440.099, Revision 1, the applicant provided its evaluation, which concluded that the Finnish Scenario is of no consequence to the AP1000 reactor design. This is because the steam generators are not relied upon to cool the RCS during a SBLOCA event. Consequently, the steam generators should not generate any significant amount of boron-free condensate via reflux condensation over an extended period of time during a SBLOCA event. The following describes the staff evaluation in detail.

During a SBLOCA, the SG functions as a "heat source" as the RCS depressurizes, rather than a "heat sink." Therefore, the differential temperature across the primary and secondary side of the generators is such that steam from the reactor will not condense on the tubes.

The AP1000 PRHR heat exchanger becomes a dominant RCS heat sink following the generation of an "S" signal during a postulated SBLOCA events. As such, the PRHR heat exchanger could become a potential source for generating a volume of unborated coolant during a SBLOCA. Such a scenario could lead to a reactivity excursion as a result of a restart of a RCP after the unborated water slug had collected in the reactor coolant loop. The applicant had determined that this scenario is not a concern for the AP1000 for the following reasons. Specifically, the AP1000 reactor coolant loop piping does not contain a loop seal, and thus, there is not a collection point for a large slug of unborated condensate to collect in the reactor coolant loop piping. During the SBLOCA event, once subcooling in the RCS is lost, steam will enter the PRHR heat exchanger, and will condense on the inside of the PRHR heat exchanger tubes. Steam condensed in the PRHR is delivered to the Loop 1 SG outlet plenum. The AP1000 loop layout does not contain an RCP crossover leg, and the PRHR condensate will drain continuously from the SG channel head into the Loop 1 cold legs, and flow into the reactor vessel.

During the SBLOCA transient, the water in the cold legs enters the downcomer, where it mixes with the highly borated safety injection flow from either the accumulators, the core makeup tanks, or both. The relatively low flow rate of fluid from the downcomer into the core, during the post RCP-trip natural circulation phase of the AP1000 SBLOCA events, enables mixing to occur in the downcomer and lower plenum. No unmixed slugs of unborated water from the PRHR can form in the downcomer and enter the core during this scenario.

The applicant performed bounding calculations that demonstrated that it was not credible to postulate that the boron concentration in the downcomer and lower plenum would be diluted to a critical boron concentration for this postulated LOCA. The conclusions from these studies showing that boron dilution from the operation of the PRHR heat exchanger would not occur, were based on demonstrating that the PRHR condensate would adequately mix with the water in the downcomer and the lower plenum, so that a critical boron concentration would not be reached.

The AP1000 uses a low boron core design with the boron concentration at beginning of Cycle (BOL) of approximately 1000 ppm. The low AP1000 core boron concentration significantly reduces the potential of the PRHR to dilute the coolant in the reactor vessel to the point of criticality. Although the AP1000 PRHR flow rate is high, the CMT flow rate, the reactor vessel downcomer and lower plenum volume for the AP1000 is also large. Taking these differences into account, the AP1000 design studies show that post-LOCA boron dilution is not a concern provided that there is good mixing in the vessel. Analysis for the AP1000 showed that mixing in the reactor vessel downcomer and lower plenum will counteract boron dilution in the core due to PRHR operation.

NUREG/IA-0004, “Thermal Mixing Tests in a Semi-annular Downcomer with Interacting Flows from Cold Legs,” dated October 1986, reported the study of the mixing of high pressure safety injection (HPI) water with primary coolant in a simulated PWR downcomer. Test #106 in NUREG/IA-0004 considers a geometry which is representative of the PRHR condensate delivery geometry into the AP1000 downcomer, namely equal flow rates of liquid entering the downcomer through two cold legs which are 90 degrees apart at the connection into the reactor vessel. The downcomer at the test facility is shorter in length (approximately 3.08 m (10 ft)) than the AP1000 dimension (approximately 6.16 m (20 ft) from the cold leg bottom to the bottom of the downcomer). The test facility therefore provides less than one-half of the mixing length available in the AP1000 downcomer. The fluid velocity in the test facility cold legs is approximately 0.14 m/s (0.45 ft/s) for the simulated High Pressure Injection (HPI) flow injection in Test #106, as indicated by the “C” series of figures in NUREG/IA-0004. This is similar to the velocity of the PRHR condensate in the cold legs for SBLOCA scenarios. Therefore, the parameters of Test #106 are such that the observed results provide meaningful insights into the mixing that occurs in the AP1000 downcomer during the SBLOCA boron dilution scenarios. The results of Test #106 illustrate that the injected plume thoroughly mixes with the resident downcomer liquid during the 3.08 m (10 ft) fall to the bottom elevation. Further support for AP1000 downcomer mixing is provided by the Test #113 results of NUREG/IA-0004.

Test #113 was run at a simulated HPI injection rate which is 3.6 times greater than that of Test #106 with a 60 degree angle between the two cold leg injection connections, as depicted in the

“D” series of photographs in NUREG/IA-0004. Test #113 results show mixing behavior in the downcomer which closely resembles that of Test #106. Test #113 indicates that the sensitivity of downcomer mixing to initial plume velocity is minor. These two tests provide compelling evidence that the diluted boron stream in the AP1000 PRHR condensate delivery scenarios is well mixed in the downcomer and that no unmixed slugs enters the lower plenum or core. These test results provide additional independent technical justification that the degree of mixing which occurs in the AP1000 downcomer during the PRHR condensate return scenarios is more than adequate to disperse a plume of diluted boron liquid. The test results support the conclusion that recriticality of the core is not of concern for SBLOCA scenarios.

Based on the information provided in the submittal of the DCD for the AP1000, and on the analysis performed by Westinghouse on behalf of this event, including the thermal tests mentioned in NUREG/1A-004, the staff find analysis in support of the possible deboration from a SBLOCA event to be acceptable.

15.2.9 Anticipated Transients Without Scram (DCD Tier 2 Section 15.8)

An ATWS event is defined as an anticipated operational occurrence (such as loss of normal feedwater, loss of condenser vacuum, or LOOP) combined with an assumed failure of the RTS to shut down the reactor. On June 26, 1984, the Commission amended the Code of Federal Regulations to include 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants" (known as the "ATWS rule"). This rule, as amended on July 6, 1984, November 6, 1986, April 3, 1989, and July 29, 1996, requires nuclear power plant facilities to reduce the likelihood of failure to shut down the reactor following anticipated transients, and to mitigate the consequences of ATWS events.

The equipment to be installed in accordance with the ATWS rule is required to be diverse from the existing RTS, and must be capable of being tested at power. This equipment is intended to provide needed diversity to reduce the potential for common-mode failures that result in an ATWS and lead to unacceptable plant conditions.

For the PWRs manufactured by Westinghouse, the basic requirements of the ATWS rule are specified in paragraph (c)(1) of 10 CFR 50.62, which states:

> Each pressurized water reactor must have equipment from sensor output to final actuation device, that is diverse from the reactor trip system, to automatically initiate the auxiliary (or emergency) feedwater and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing reactor trip system.

The AP1000 design includes a control-grade diverse actuation system (DAS) to provide an alternate turbine trip signal, and an alternate actuation signal of the PRHR system for decay heat removal, which are separate and diverse from the safety-grade reactor trip system and

PRHR normal actuation signals. The DAS also provides a diverse scram function. The staff's review and acceptance of the applicant's DAS design is discussed in Section 7.7 of this report.

The AP1000 design relies on the PRHR in lieu of an auxiliary or emergency feedwater system as its safety-related method of removing decay heat. In its letter, "AP1000 Request for Exemption," DCP/NRC1534, dated December 3, 2002, the applicant submitted a request for exemption from the part of the ATWS regulation, 10 CFR 50.62(c)(1), that requires auxiliary or emergency feedwater as an alternate system for decay heat removal during an ATWS event. The staff concludes that the applicant has met the intent of the ATWS rule by relying on the PRHR system to remove the decay heat, and meets the underlying purpose of the rule. Therefore, the Commission has determined that the special circumstances described in 10 CFR 50.12(a)(2)(ii) exist in that the requirement for an auxiliary or emergency feedwater system is not necessary to achieve the underlying purpose of 10 CFR 50.62(c)(1), because the applicant adopted acceptable alternatives that accomplish the intent of this regulation, and the exemption is authorized by law, will not present an undue risk to public health and safety, and is consistent with the common defense and security.

The applicant provides the results of an ATWS analysis in DCD Tier 2 Section 19A.4 (Revision 0) for the staff to review. In its analysis, the applicant uses a complete loss of normal feedwater (LONF) event as an initial event for the ATWS analysis because the LONF event was previously established as the limiting case (i.e., produced the maximum RCS pressure) for the AP600.

The applicant performs the ATWS analysis with the LOFTRAN code. The AP1000 DAS is credited to function in the analysis for ATWS cases. Specifically, the DAS is credited to actuate a turbine trip and the PRHR when the low wide-range SG level signal is generated. Two LONF cases are analyzed: one for the equilibrium cycle core (ECC) conditions and one for the first cycle core (FCC) conditions. For the ECC case, the moderator temperature coefficient (MTC) used is -12.5 pcm/°F, which is the least negative MTC at any time in an ECC condition. The analysis shows that the peak RCS pressure is about 20.68 mPa (3000 psia), less than the limit of 22.06 mPa (3200 psia). For the FCC case, the LONF analysis shows that with the MTC of -10 pcm/°F, the calculated peak RCS reaches to about 22.06 mPa (3200 psia). The MTC used in the analysis for the FCC case is less than 60 percent of the cycle time. Therefore, the unfavorable exposure time (UET) for the FCC case is 40 percent. The UET is the time during the fuel cycles when the reactivity feedback is not sufficient to maintain pressure under 22.06 mPa (3200 psia) for a given reactor state. During the review, the staff requested in a follow-up question to RAI 440.014 the applicant provide additional justification to show that the LONF analysis is the worst case and the selection of the MTC used in the analysis is acceptable with respect to the acceptable UET limit.

In response, the applicant performs a probabilistic risk assessment (PRA) evaluation to identify the frequency of the anticipated transients in DCD Tier 2 Chapter 15 and specifies the most risk-significant ATWS scenarios for the AP1000. Based on the results of the applicant's PRA evaluation, the applicant identifies the most risk-significant ATWS scenarios which make up more than 95 percent of the AP1000 ATWS initiating event frequency. The applicant performs ATWS analyses on the most risk-significant ATWS cases that, based on the results of DCD

Tier 2 Chapter 15 non-LOCA analysis for the AP1000, may result in a significant pressure increase during the transients. The following cases are analyzed in detailed by the applicant to identify the scenario that results in the least margin to the reactor coolant pressure boundary limit for the AP1000:

- Turbine trip without feedwater system operable with turbine bypass system operable
- Turbine trip with feedwater system operable with turbine bypass system operable
- Turbine trip without feedwater system operable without turbine bypass system operable
- Loss of normal feedwater event with turbine bypass system operable
- Loss of normal feedwater event, without spray system, without turbine bypass system operable
- Loss of normal feedwater event with turbine bypass system operable, more realistic SG heat transfer model
- Complete loss of forced coolant flow with main feedwater (MFW) system operable with turbine bypass system operable
- Complete loss of forced coolant flow with MFW system operable without turbine bypass system operable
- Complete loss of forced coolant flow induced by the loss of ac power, MFW system not operable, no steam dump operable, turbine trip at the initiation of the transient

The results of the ATWS analysis for the above cases confirms that for the AP1000, the limiting case is the loss of normal feedwater event with the turbine bypass operable, resulting in the highest peak RCS pressure.

In addressing the second concern in RAI 440.014, related to the acceptability of the MTC value used in the ATWS analysis for the limiting case, the applicant has revised the DAS actuation logic to improve its capability of accident mitigation in response to an ATWS event. In addition to actuation of the PRHR and the turbine trip, the new logic actuates the CMT and RCP trip on the low wide-range SG. Together with implementation of a new DAS logic, an additional change has been implemented in the plant control system (PLS) such that the PLS isolates the steam dump system whenever the SG level drops below the low SG water level wide-range setpoint. The description of the new DAS logic is included in DCD Tier 2 Section 7.1.1.

The review and acceptance of the new logic are discussed in Section 7.7.1 of this report. The applicant performed an ATWS analysis for the loss of normal feedwater event for the AP1000, with the new DAS ATWS protection logic assuming a MTC of -5 pcm/°F. The value of the MTC envelops 100 percent of the AP1000 core life. The results of the ATWS show that for the limiting case, the loss of normal feedwater event, the maximum calculated RCS pressure is 19.42 MPa (2,818 psia) which is within the acceptance limit of 22.06 MPa (3,200 psia). The limiting ATWS case demonstrates that the UET (i.e., the time during the fuel cycles when the reactivity feedback is not sufficient to maintain pressure under 22.06 MPa (3,200 psia) for a given reactor state) does not exist for the AP1000. The information of the new ATWS analysis discussed in this section is included in the RAI 440.014, Revision 1 and DCD Tier 2 Section 19A.4 (Revision 1). Since (1) the previously NRC-approved LOFTRAN code is used for the ATWS analysis, (2) the limiting ATWS case is identified based on the results of the actual ATWS analysis discussed in this section, (3) the value of the MTC used in the limiting case

analysis envelops 100 percent of the AP1000 fuel cycles, and (4) the calculated peak RCS pressure for the limiting case is within the pressure limit acceptance criterion, the staff concludes that the ATWS analysis is acceptable.

15.2.10 Conclusions

The staff has reviewed the safety analyses of the design basis transients and accidents described DCD Tier 2 Chapter 15 for the AP1000 design. Based on the evaluation discussed above, the staff concluded that the AP1000 design meets the acceptable criteria of these transients and accidents.

As discussed in Section 15.2.9 of this report, the staff concluded that the application's request for exemption to the ATWS rule of 10 CFR 50.62 is acceptable. Specifically, the exemption request applies to 10 CFR 50.62(c)(1), which requires auxiliary or emergency feedwater as an alternate system for decay heat removal during an ATWS event. The AP1000 design relies on the PRHR in lieu of an auxiliary or emergency feedwater system as its safety-related method of removing decay heat. The staff concludes that the AP1000 design meets the underlying purpose of the ATWS rule by relying on the PRHR system to remove the decay heat. In accordance with 10 CFR 50.12(a)(2)(ii) for the AP1000 design, the requirement for an EFS is not necessary to acheive the underlying purpose of 10 CFR 50.62(c)(1). Therefore, the exemption request is acceptable.

The staff also identified the following open items.

- Open Item 15.1.5-1 (COL Action Item 15.1.5-1) regarding SG Mid-deck plate induced level measurement uncertainty:

  The COL applicant should evaluate the validity of the safety analysis documented in the AP1000 DCD using plant-specific setpoints and instrument uncertainties, including the SG mid-deck plate level measurement uncertainties. The COL applicants should submit in the plant specific applications the evaluation results. The applicant should include this COL Action Item in the DCD.

- Open Item 15.2.7-1 regarding the core void distribution during LTC:

  The information provided regarding core void distribution was not sufficiently detailed to draw the conclusion that adiabatic heating would be avoided. The applicant should provide more detailed information of the axial void distribution during LTC and show that the possibility of adiabatic fuel heating is excluded.

15.3 Radiological Consequences of Accidents

In DCD Tier 2 Chapter 15, the applicant performed radiological consequence assessments of the following seven reactor design-basis accidents (DBAs) using the hypothetical set of atmospheric relative concentration (dispersion) values (χ/Q values) provided in DCD Tier 2 Table 15A-5. Given all other aspects of the design are fixed, these χ/Q values help determine

the required minimum distances to the exclusion area boundary (EAB) and the low-population zone (LPZ) for a given site in order to provide reasonable assurance that the radiological consequences of a DBA will be within the dose limits specified in 10 CFR 50.34(a)(1)(ii)(D). The analyzed DBAs are:

- main steam line break outside containment (DCD Tier 2 Section 15.1.5)
- RCP shaft seizure (locked rotor) (DCD Tier 2 Section 15.3.3)
- rod cluster control assembly ejection (DCD Tier 2 Section 15.4.8)
- failure of small lines carrying primary coolant outside containment (DCD Tier 2 Section 15.6.2)
- SGTR (DCD Tier 2 Section 15.6.3)
- LOCA (DCD Tier 2 Section 15.6.5)
- fuel handling accident (DCD Tier 2 Section 15.7.4)

In DCD Tier 2 Chapter 15, the applicant concludes that the AP1000 design will provide reasonable assurance that the radiological consequences resulting from any of the above DBAs will be within the offsite dose criteria specified in 10 CFR 50.34(a)(1)(ii)(D) of 0.25 Sv (25 rem) total effective dose equivalent (TEDE) and the control room operator dose criterion specified in GDC 19 of Appendix A to 10 CFR Part 50 of 0.05 Sv (5 rem) TEDE. The applicant reached this conclusion by:

- using reactor accident source terms based on NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," and RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,”
- relying on natural deposition of fission-product aerosol within the containment,
- controlling the pH of the water in the containment to prevent iodine evolution, and
- using a set of hypothetical atmospheric dispersion factor (χ/Q) values.

The χ/Q values are the relative atmospheric concentrations of radiological releases at the receptor point in terms of the rate of radioactivity release. In lieu of site-specific meteorological data, the applicant provided a reference set of χ/Q values for the AP1000 design using the meteorological data that is representative of a 60th to 70th percentile of U.S. operating nuclear power plant sites, for offsite dispersion. The AP1000 hypothetical χ/Q values are listed in DCD Tier 2 Tables 2-1 and 15A-5.

Regulatory Evaluation

The staff evaluated the radiological consequences of DBAs against the dose criteria specified in 10 CFR 50.34(a)(1)(ii)(D) of 0.25 Sv (25 rem) TEDE at the EAB for any 2-hour period following the onset of the postulated fission product release, and 0.25 Sv (25 rem) TEDE at the outer boundary of the low population zone for the duration of exposure to the release cloud. The staff used a criterion of 0.05 Sv (5 rem) TEDE for evaluating the radiological consequences from DBAs in the control room of the AP1000 design, pursuant to GDC 19 of Appendix A to 10 CFR Part 50. The staff used applicable guidance in SRP Section 15.0.1, “Radiological Consequence Analyses Using Alternative Source Terms,” and RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” in its review of the AP1000 design basis accident radiological consequences analyses. RG 1.183 was written to provide guidance on radiological consequences analyses to licensees of operating power reactors that choose to implement an alternative source term pursuant to 10 CFR 50.67, which has the same regulatory dose criteria specified in 10 CFR 50.34(a)(1)(ii)(D) (0.25 Sv (25 rem) TEDE) and GDC 19 (0.05 Sv (5 rem) TEDE). Although RG 1.183 was written to apply to the current operating power reactors, its guidance on radiological acceptance criteria, formulation of the source term and DBA modeling is useful in the review of the AP1000 design because it is an advanced PWR.

Technical Evaluation

The staff reviewed the radiological consequence analyses performed by Westinghouse using the hypothetical χ/Q values in DCD Tier 2 Table 15A-5, and finds that the radiological consequences calculated by Westinghouse meet the relevant dose acceptance criteria stated above. To verify the Westinghouse analyses, the staff performed independent radiological calculations for the above DBAs using the hypothetical χ/Q values provided by the applicant and the computer code described in Supplement 2 to NUREG/CR-6604, “RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation.” The staff’s findings are described in the following sections.

Accident Source Terms

In SECY-94-302, "Source Term-Related Technical and Licensing Issues Relating to Evolutionary and Passive Light-Water-Reactor Designs," dated December 19, 1994, the staff proposed to use only the "coolant," "gap," and "early in-vessel" releases from NUREG-1465 for the radiological consequence assessments of DBAs for the passive advanced light water reactor (ALWR) designs. These source terms encompass a broad range of accident scenarios, including significant levels of core damage with the core remaining in the vessel. These would be the most severe scenarios from which the plant could be expected to return to a safe-shutdown condition. The revised source terms in NUREG-1465 are to be applied conservatively in evaluating DBAs in conjunction with conservative assumptions in calculating doses, such as adverse meteorology. Application to severe accidents may use more realistic assumptions.

The staff considered the inclusion of the "ex-vessel" and the "late in-vessel" source terms to be unduly conservative for DBA purposes. Such releases would only result from core damage accidents with vessel failure and core-concrete interactions. For passive ALWRs, the estimated frequencies of such scenarios are low enough that they need not be considered credible for the purpose of meeting 10 CFR 50.34. In SECY-94-302 (December 1994), the Commission approved the staff-recommended technical positions to use only the coolant, gap, and early in-vessel releases from NUREG-1465 for the radiological consequence assessments of DBAs for the passive ALWR designs.

RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” was issued in July 2000 to provide guidance to licensees of operating power reactors on acceptable applications of alternative source terms pursuant to 10 CFR 50.67. This RG establishes an acceptable alternative source term based on insights from NUREG-1465, and also establishes significant attributes of other alternative source terms that may be found acceptable by the NRC staff for operating light water reactors (LWRs). It also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted alternative source term for operating power reactors. The applicant followed applicable guidance in RG 1.183 for PWRs.

Post-Accident Containment Water Chemistry Management

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

NUREG-1465 specifies that, after an accident, iodine entering the containment from the reactor core is composed of at least 95-percent cesium iodide (CsI) with the remaining 5 percent being elemental iodine and a small amount of hydriodic acid. However, about 3 percent of elemental iodine in contact with some organic compounds will produce organic iodides. Therefore, the iodine in the containment will consist of 95-percent particulate iodine as CsI, 4.85 elemental iodine (I_2), and 0.15-percent organic iodine. The composition of the iodine in AP1000 is consistent with the composition specified in NUREG-1465.

Iodine in the form of cesium iodide is soluble in the containment water. However, some of it may be converted into the elemental form, which is considerably less soluble and will be released into the containment atmosphere. The released radioactive iodine may leak out of the containment and contribute to outside radiation doses. In order to minimize formation of the elemental iodine, the pH of the containment water should be kept basic. Basic pH will also prevent stress corrosion cracking of the stainless steel components.

In AP1000, the pH of the containment water will be between 7 and 9.5. The range is the acceptable range specified in Branch Technical Position MTEB 6-1 of the Standard Review Plan. It will be maintained in this range by a predetermined amount of trisodium phosphate (TSP) stored in the stainless steel baskets situated on the containment floor. After a LOCA, when the containment flooding water reaches the level of the baskets, TSP will dissolve and the solution of TSP will exercise buffering action and maintain sump water pH in the required range.

The pH of the containment sump water after a LOCA is determined by acidic and basic chemical species released to the containment from different sources in the plant. The most significant effect on reducing containment water pH is produced by boric acid. Westinghouse identified the following sources of boric acid: RCS, IRWST, CMTs, Accumulators, CVS BAT, and spent fuel pool cooling system (SFS) Cask Loading Pit. There was no need to include normal operating RCS leakage because in the AP1000 plant design such leakage would quickly drain to the waste sump and be pumped out of the containment. Other sources of chemical species that are formed in the containment during 30 days following a core damage accident are: hydrochloric acid produced by radiolytic decomposition of electric cable jackets, nitric acid produced by radiolytic formation from air dissolved in the sump water, and cesium hydroxide from the damaged core. Cesium hydroxide, being a strong base, will contribute to the increase of pH. The applicant has determined that in order to maintain pH in the range of 7 to 9.5, 12,492 kg (27,540 pounds) of TSP is required to be stored in the baskets in the containment sump. The amount of TSP required to neutralize boric acid is 7,503 kg (16,540 pounds). The rest is needed to neutralize other acids existing in the containment and to provide a 35-percent safety margin which includes 10 percent needed to cover possible long term degradation of TSP.

By performing independent verifications, based on the information provided in the response to RAI 281.004, the staff has confirmed the adequacy of the post-accident management of water chemistry in the AP1000 plant design because the amount of TSP stored in the containment sump will ensure a basic environment that will minimize release of radioactive iodine to the outside and will prevent stress corrosion cracking of the stainless steel components exposed to the containment water. The staff concludes, therefore, that the AP1000 plant design meets the requirements of GDC 41 and GDC 4.

Aerosol Removal Mechanisms

An active containment atmosphere cleanup system has not been provided for the AP1000 design. Reliance is placed on natural aerosol removal processes for deposition in the containment such as the sedimentation mechanism of gravitational settling and the diffusion mechanisms of diffusiophoresis and thermophoresis for plateout on containment surfaces. Discussion of the removal of airborne activity from the containment atmosphere can be found in DCD Tier 2 Appendix 15B. The applicant provided a containment spray system for accident management following a severe accident as part of the AP1000 fire protection system design. (See Section 19.2.3.3.9 of this report.) The containment spray system design is not safety-related, and is not intended to be used during or following a DBA. Therefore, no credit for mitigation of radiological releases following a DBA is given for the containment spray system in radiological consequence assessments.

The determination of aerosol removal rates is simplified for a containment shown to have a well-mixed atmosphere. The AP1000 design relies on natural circulation currents enhanced by the passive containment cooling system to inhibit stratification of the containment atmosphere. The physical mechanisms of natural circulation mixing that occur in the AP1000 are discussed in DCD Tier 2 Appendix 6A. See Section 6.2.5 of this report for the staff's discussion of natural circulation within the AP1000 containment.

In DCD Tier 2 Table 15B-1, the applicant provides aerosol removal coefficients starting at the onset of gap release through the first 24 hours into a DBA. The values range between 0.43 to 0.72 per hour. These are the same values the applicant used for the AP600 design, and were calculated using the plant parameters for the AP600, not the AP1000. As discussed in DCD Tier 2 Appendix 15B, the applicant considers the use of the AP600 aerosol removal coefficients to be conservative for the AP1000 design. The greater containment volume (approximately 20 percent greater) for the AP1000 would result in a reduction in airborne aerosol concentration which, with all other things being equal, would result in a reduction in the aerosol removal coefficient. This reduction is offset by a number of other differences between the two designs, as identified in DCD Tier 2 Section 15B.2.4:

- Increased heat removal area compared to the AP600 design

- Greater fission product activity released to the containment atmosphere due to the higher power of the AP1000 core (thus increasing airborne aerosol concentration)

- Increased decay heat to be removed through the containment shell

- Increased post-LOCA temperature and pressure inside containment resulting in higher heat transfer rates

The staff has not completed its evaluation of the applicability of the AP600 aerosol removal coefficients to the AP1000 design. The staff will evaluate the impact of the differences in the AP1000 design as compared to the AP600 on the modeling of aerosol removal and will perform independent analyses of the estimated aerosol removal rates. Upon resolution of issues with the determination of aerosol removal rates in containment, as discussed in RAIs 470.009 and 470.011, the staff will complete its evaluation of the bounding accident sequence and the aerosol behavior and removal rates corresponding to the selected bounding accident sequence in the containment following a DBA. This is Open Item 15.3-1.

Hypothetical Atmospheric Dispersion Factors

Because no specific site is associated with the AP1000 plant, Westinghouse defined the offsite boundaries only in terms of various hypothetical atmospheric relative concentration (χ/Q) values at fixed EAB and LPZ distances. The hypothetical reference χ/Q values used in the radiological consequences analyses for the AP1000 design are listed in DCD Table 15A-5. The staff's discussion of the hypothetical atmospheric dispersion factors is located in Section 2.3.4 of this report. The staff will perform an independent assessment of short-term ($\leq$ 30 days) atmospheric dispersion factors for potential accident consequence analyses on a site-specific

basis for a COL application that references the AP1000 design. If site-specific atmospheric dispersion factors are greater than the reference values (e.g., poorer dispersion characteristics) used in this evaluation, a COL applicant may have to consider compensatory measures, such as increasing the size of the site or providing additional engineered safety feature systems, to meet the relevant dose limits set forth in 10 CFR 50.34 and GDC 19.

Staff review of the control room atmospheric dispersion factors is not complete (see Section 2.3.4 of this report). Pending resolution of staff's concerns with the hypothetical reference control room χ/Q values, review of the control room habitability radiological consequences analysis is also incomplete for each of the following design basis accidents. Inputs other than the assumed χ/Q values have been found acceptable. Using the control room χ/Q values provided in the AP1000 DCD, the staff has also preliminarily confirmed Westinghouse's control room doses for accidents other than the LOCA. Upon completion of the staff's review of the applicant's hypothetical reference control room χ/Q values, the staff will complete review of the control room habitability radiological consequences analyses. This is Open Item 15.3-2.

15.3.1 Radiological Consequences of a Main Steamline Break Outside Containment

Both the staff and Westinghouse have evaluated the radiological consequences of a postulated steam line break accident occurring outside of the containment and upstream of the main steam isolation valves. The applicant submitted a radiological analysis for the MSLB accident in DCD Tier 2 Section 15.1.5.4. The applicant analyzed this hypothetical accident using

(1) 1,893 liters per day (500 gallons per day) of primary-to-secondary leakage through any one SG, as specified in the AP1000 TS

(2) discharge of the entire mass of secondary water from one affected SG to the environment with no iodine partitioning

The applicant also considered a coincident loss of spent fuel pool cooling capability. Discussion of the staff's review of the radiological consequences of spent fuel pool boiling is presented in Section 15.3.9 of this report.

The staff has reviewed the applicant's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable, and that the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria.

To verify the applicant's assessment, the staff performed an independent radiological consequence calculation for three scenarios for the MSLB accident. For Case 1, the most reactive control rod was assumed to be stuck in the fully withdrawn position. The applicant indicates, and the staff agrees, that no DNB is expected to occur; therefore, no fuel-cladding failure was assumed in the calculation. With no additional fuel failures occurring, Case 1 becomes identical to Case 2 (discussed below), and no radiological consequences are presented for Case 1.

For Case 2, the staff assumed that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by the main steam line break accident. Before the accident, the AP1000 reactor was assumed to be operating at the AP1000 TS equilibrium limit of 37 kBq/gm (1.0 µCi/gm) dose equivalent iodine-131 (DEI-131) in the primary coolant. The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate results in an increasing concentration in the primary coolant during the course of the accident. For Case 3, the staff assumed that previous reactor operation had resulted in a primary coolant iodine concentration equal to the maximum instantaneous AP1000 TS limit of 2.2 MBq/gm (60 µCi/gm) DEI-131.

The major parameters and assumptions used by the staff for the MSLB accident are provided in Table 15.3-2 of this report, and the results of the staff's radiological consequence analyses for the EAB, LPZ, and control room are provided in Table 15.3-1 of this report. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant.

The staff concludes that the AP1000 design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated MSLB accident with accident-induced iodine spiking and coincident with the loss of spent fuel pool cooling capability will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criteria set forth in 10 CFR 50.34.

The staff concludes that the AP1000 design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated MSLB accident with the reactor coolant at the TS maximum value of 2.2 MBq/gm (60 µCi/gm) DEI-131 and coincident with the loss of spent fuel pool cooling capability will not exceed the dose criteria set forth in 10 CFR 50.34 (0.25 Sv (25 rem) TEDE).

As discussed in Open Item 15.3-2, the staff has not completed its review of the applicant's control room habitability radiological consequences analyses. The staff has not made a determination as to whether the dose criterion of 0.05 Sv (5 rem) TEDE in GDC-19 is met for the MSLB.

15.3.2 Radiological Consequences of a Reactor Primary Coolant Pump Seizure (Locked Rotor)

The reactor primary coolant pump seizure accident is caused by an instantaneous seizure of an RCP rotor rapidly reducing the primary coolant flow through the affected reactor coolant loop leading to a reactor trip on a low-flow signal. Westinghouse analyzed this hypothetical accident assuming that 16 percent of the fuel elements will experience cladding failure, releasing the entire fission product inventory in the fuel-cladding gap of these elements to the reactor coolant. Activity released to the primary coolant is carried to the secondary coolant by the maximum allowable 3785 liters/day (1000 gallons/day) of primary-to-secondary leakage through two SGs as specified in the AP1000 TS. Activity is released to the environment via the steam line safety

valves or the power-operated relief valves. The applicant submitted a radiological analysis for the reactor primary coolant pump seizure accident in DCD Tier 2 Section 15.3.3 of the DCD.

The applicant also considered a coincident loss of spent fuel pool cooling capability. Discussion of the staff's review of the radiological consequences of spent fuel pool boiling is presented in Section 15.3.9 of this report.

The staff has reviewed the applicant's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable, and that the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria.

To verify the applicant's assessment, the staff performed independent radiological consequence calculations for the reactor primary coolant pump seizure accident using RG 1.183 source terms. The major parameters and assumptions used by the staff are provided in Table 15.3-3 of this report, and the results of the staff's radiological consequence analyses are provided in Table 15.3-1 of this report. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant.

The staff concludes that the AP1000 design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated reactor primary coolant pump seizure accident coincident with the loss of spent fuel pool cooling capability will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criteria set forth in 10 CFR 50.34.

As discussed in Open Item 15.3-2, the staff has not completed its review of the applicant's control room habitability radiological consequences analyses. The staff has not made a determination as to whether the dose criterion of 0.05 Sv (5 rem) TEDE in GDC-19 is met for the primary coolant pump seizure accident.

15.3.3 Radiological Consequences of Rod Cluster Control Assembly Ejection

The mechanical failure of a control rod mechanism pressure housing is postulated to result in the ejection of a RCCA and drive shaft. Because of the resultant opening in the pressure vessel, primary coolant is lost to the containment with concurrent rapid depressurization of the reactor pressure vessel. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The applicant has assumed that 10 percent of the fuel elements will experience cladding failure, releasing the entire fission product inventory in the fuel-cladding gap of these elements. In addition, the applicant assumed that 0.25 percent of the fuel rods may experience fuel melting. The applicant performed its calculations to obtain these parameters using the guidelines provided in RG 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for PWRs;" therefore, the staff finds these assumptions to be acceptable. The applicant submitted a radiological consequence analysis for the control element assembly ejection accident in DCD Tier 2 Section 15.4.8.

The applicant also considered a coincident loss of spent fuel pool cooling capability. Discussion of the staff's review of the radiological consequences of spent fuel pool boiling is presented in Section 15.3.9 of this report.

The staff has reviewed the applicant's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable and that the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria.

The applicant assumed that the release of fission products to the environment may occur via either of two pathways. The first pathway involves a release of primary coolant to the containment, which is then assumed to leak to the environment at the design leak rate of the containment. In the second pathway, fission products would reach the secondary coolant via the SGs with a maximum total allowable primary-to-secondary leak rate of 3785 liters/day (1000 gallons/day) as specified in the AP1000 TS. For both pathways, the applicant assumed that the AP1000 reactor was operating at its TS instantaneous primary coolant limit of 2.2 MBq/gm (60 µCi/gm) for DEI-131.

To verify the applicant's assessment, the staff performed independent radiological consequence calculations for the same two pathways as described above for the RCCA ejection accident using RG 1.183 source terms. The major parameters and assumptions used by the staff are provided in Table 15.3-4 of this report, and the results of the staff's radiological consequence analyses are provided in Table 15.3-1 of this report. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant.

The staff concludes that the AP1000 design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated rod cluster control assembly ejection accident coincident with the loss of spent fuel pool cooling capability will be well within the dose criteria set forth in 10 CFR 50.34 (i.e., 25 percent or 0.063 Sv (6.3 rem) TEDE).

As discussed in Open Item 15.3-2, the staff has not completed its review of the applicant's control room habitability radiological consequences analyses. The staff has not made a determination as to whether the dose criterion of 0.05 Sv (5 rem) TEDE in GDC-19 is met for the rod cluster control assembly ejection accident.

15.3.4 Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment

GDC 55 contains a provision to ensure isolation of all pipes that are part of the RCPB and penetrate the containment building. GDC 55 also provides that small-diameter pipes that must be continuously connected to the primary coolant system in order to perform necessary functions may be acceptable based on some other defined bases. For these lines, methods of

mitigating the consequences of a rupture are necessary because the lines cannot be automatically isolated. For the AP1000 design, there are two small lines in this category:

- the reactor coolant system sample line
- the discharge line from the CVS to the liquid radwaste system

No instrument lines carry primary coolant outside containment in the AP1000 design.

When excess primary coolant inventory is generated as a result of boron dilution operations, the CVS purification flow is diverted out of containment to the liquid radwaste system. Before passing outside containment, the flow stream passes through the CVS heat exchangers and mixed bed demineralizer. The flow leaving the containment will be at temperature of less than 60 °C (140 °F) and has been processed by the demineralizer. The flow from a postulated break in this line is limited to the CVS purification normal flow rate of 379 liters/minute (100 gpm). Considering the low temperature of the break flow and the reduced iodine activity because of demineralization for the postulated break in the discharge line from the CVS to the liquid radwaste system, the applicant proposed in DCD Tier 2 Section 15.6.2, and the staff accepted, that the postulated break in the RCS sample line is the more limiting event for the radiological consequence assessment.

The RCS sample line includes a flow restrictor at the point of sample to limit the break flow to less than 492 liters/minute (130 gpm). Because the sample line isolation valves are only open when sampling is ongoing and there are multiple indications that a break has occurred in the sample line, the applicant assumed, and the staff accepted, that the break flow isolation time will be less than 30 minutes. The fluid escaping the break is assumed by the applicant to be at the equilibrium primary coolant iodine concentration limits in the AP1000 TS with an assumed accident initiated iodine spike that increases the rate of iodine release from the fuel into the coolant by a factor of 500. The staff finds this to be acceptable and in agreement with guidance on assumptions for radioactivity released from a small line break found in SRP 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment." The applicant submitted a radiological analysis for a small line failure in DCD Tier 2 Section 15.6.2.

The applicant also considered a coincident loss of spent fuel pool cooling capability. Discussion of the staff's review of the radiological consequences of spent fuel pool boiling is presented in Section 15.3.9 of this report.

The staff has reviewed the the applicant analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria. To verify the the applicant assessment, the staff performed independent radiological consequence calculations for a postulated small line break accident. The staff assumed that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by the small line break accident. Before the postulated accident, the AP1000 reactor was assumed to be operating at the AP1000 TS equilibrium concentration limit of 37 kBq/gm (1.0 µCi/gm) DEI-131 in the primary coolant.

The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 500. This increase in the release rate results in an increasing iodine concentration in the primary coolant during the course of the accident.

The major parameters and assumptions used by the staff are provided in Table 15.3-5 of this report, and the results of the staff's radiological consequence analyses are provided in Table 15.3-1 of this report. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant.

The staff concludes that the AP1000 design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated small line break accident coincident with the loss of spent fuel pool cooling capability will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criteria set forth in 10 CFR 50.34.

As discussed in Open Item 15.3-2, the staff has not completed its review of the applicant's control room habitability radiological consequences analyses. The staff has not made a determination as to whether the dose criterion of 0.05 Sv (5 rem) TEDE in GDC-19 is met for the small line break accident.

15.3.5 Radiological Consequences of a Steam Generator Tube Rupture

The applicant has evaluated the radiological consequences of a postulated SGTR accident and provided a radiological consequence analysis for the accident in DCD Tier 2 Section 15.6.3. The staff has reviewed the applicant's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable, and the radiological consequences calculated by Westinghouse meet the relevant dose acceptance criteria.

The applicant also considered a coincident loss of spent fuel pool cooling capability. Discussion of the staff's review of the radiological consequences of spent fuel pool boiling is presented in Section 15.3.9 of this report.

To verify the applicant's assessments, the staff performed independent radiological consequence calculations for two scenarios for the SGTR accident. For Case 1, the staff assumed that a temporary increase in the primary coolant iodine concentration (iodine spike) occurred as a result of the power/pressure transient caused by the SGTR. Before the postulated accident, the AP1000 reactor was assumed to be operating at the AP1000 TS equilibrium iodine concentration limit of 37 kBq/gm (1.0 µCi/gm) DEI-131 in the primary coolant. The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 335. This increase in the release rate results in an increasing iodine concentration in the primary coolant during the course of the accident.

For Case 2, the staff assumed that previous reactor operation had resulted in a primary coolant concentration equal to the maximum instantaneous concentration limit of 2.2 MBq/gm (60 µCi/gm) DEI-131 specified in the AP1000 TS. The major parameters and assumptions used by the staff are provided in Table 15.3-6 of this report, and the results of the staff's radiological

consequence analyses for the exclusion area boundary and low population zone and for the control room are provided in Table 15.3-1. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant.

The staff concludes that the AP1000 design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated SGTR accident with accident-induced iodine spiking and coincident with the loss of spent fuel pool cooling capability will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criteria set forth in 10 CFR 50.34.

The staff concludes that the AP1000 design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated SGTR accident with the reactor coolant at the TS maximum value of 2.2 MBq/gm (60 µCi/gm) DEI-131 and coincident with the loss of spent fuel pool cooling capability will not exceed the dose criteria set forth in 10 CFR 50.34 (0.25 Sv (25 rem) TEDE).

As discussed in Open Item 15.3-2, the staff has not completed its review of the applicant’s control room habitability radiological consequences analyses. The staff has not made a determination as to whether the dose criterion of 0.05 Sv (5 rem) TEDE in GDC-19 is met for the SGTR.

15.3.6 Radiological Consequences of LOCAs

In DCD Tier 2 Section 15.6.5, the applicant analyzed a hypothetical design-basis LOCA. The applicant concludes that certain bounding sets of atmospheric relative concentration values specified in DCD Tier 2 Section 2.3, in conjunction of the use of natural deposition of fission product aerosol within the containment and controlling the pH of the water in the containment to prevent iodine evolution, are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated design-basis LOCA will be within the relevant dose criteria established in 10 CFR 50.34 and in GDC 19.

All of the fission product releases due to the LOCA are the result of containment leakage. The AP1000 design does not have ESF systems outside of the containment; therefore, no leakage from the ESF systems is considered for the radiological consequence analyses. The containment was assumed to leak at its design leak rate of 0.1 weight percent per day for the entire duration of the accident (30 days). The AP1000 design provides neither an ESF filtration (e.g., charcoal adsorbers) nor a safety-related containment spray system.

The applicant also considered a coincident loss of spent fuel pool cooling capability. Discussion of the staff’s review of the radiological consequences of spent fuel pool boiling is presented in Section 15.3.9 of this report.

The staff has not completed its evaluation of the applicant’s assumptions on aerosol removal in containment, as discussed in RAIs 470.009 and 470.011. To verify the applicant’s assessment, the staff will perform independent radiological consequence calculations for a postulated

design-basis LOCA coincident with the loss of spent fuel pool cooling capability once these issues are resolved. This is Open Item 15.3.6-1.

15.3.7 Radiological Consequences of Fuel Handling Accident

In DCD Tier 2 Section 15.7.4, the applicant presented its analyses of the radiological consequences of a postulated fuel handling accident (FHA). For the AP1000 design, an FHA can be postulated to occur either inside containment or in the fuel handling area inside the auxiliary building. If the FHA occurs in the containment, the release of fission products can be terminated by closure of the containment purge lines based on the detection of high airborne radioactivity. The applicant assumed, in accordance with guidance in RG 1.183, that fission products are directly released to the environment within a 2-hour period without credit for any iodine removal processes.

For the FHA, the applicant assumed a single fuel assembly that has undergone 100 hours of decay time is dropped such that the activity in the gap of every rod in the dropped assembly is released. The kinetic energy of the falling fuel assembly is assumed to break open the maximum possible number of fuel rods using perfect mechanical efficiency. Instantaneous release of noble gases and radioiodine vapor from the gaps of the broken rods (8 percent of I-131, 10 percent of Kr-85 and 5 percent of other iodine and noble gas inventories in the fuel rod) is assumed to occur, with the released gases bubbling up through the fuel pool water (with an effective decontamination factor of 200 for total iodine). These gap fractions are in agreement with RG 1.183 guidance. The applicant assumed that iodine in the particulate form is not volatile, and therefore, not released. In accordance with RG 1.183 guidance, the applicant assumed that the particulate CsI is instantaneously converted to the elemental form of iodine when it is released from the fuel into the low pH pool water.

The applicant also considered a coincident loss of spent fuel pool cooling capability. Discussion of the staff's review of the radiological consequences of spent fuel pool boiling is presented in Section 15.3.9 of this report.

The staff has reviewed the applicant's analysis and finds that the calculational methods used for the radiological consequence assessment are acceptable, and the radiological consequences calculated by the applicant meet the relevant dose acceptance criteria.

To verify the applicant's assessments, the staff performed independent radiological consequence calculations for the fuel handling accident occurring 100 hours after shutdown, coincident with a loss of the spent fuel pool cooling capability. The major parameters and assumptions used by the staff are provided in Table 15.3-10 of this report, and the results of the staff's radiological consequence analyses are provided in Table 15.3-1 of this report. The offsite radiological consequences calculated by the staff are consistent with those calculated by Westinghouse.

The staff concludes that the AP1000 design, as bounded by the atmospheric relative concentrations proposed by the applicant, will provide reasonable assurance that the radiological consequences of a postulated fuel handling accident at 100 hours post-shutdown

with the loss of spent fuel pool cooling capability will be well within the dose criteria set forth in 10 CFR 50.34 (i.e., 25 percent or 0.063 Sv (6.3 rem) TEDE).

As discussed in Open Item 15.3-2, the staff has not completed its review of the applicant's control room habitability radiological consequences analyses. The staff has not made a determination as to whether the dose criterion of 0.05 Sv (5 rem) TEDE in GDC-19 is met for the FHA.

In RAI 630.052, the NRC staff asked the applicant for justification to not include a TS LCO for a decay time limit related to the assumption used in the radiological consequences analysis of the FHA. The applicant, in response to the RAI, stated that they had performed a sensitivity study in which the FHA was assumed to occur 24 hours after shutdown and the resulting doses remain below 10 CFR 50.34 guidelines. The applicant updated DCD to include a paragraph discussing an evaluation of the FHA performed assuming a decay time of 24 hours. The applicant asserts that a decay time LCO is not needed because the evaluation of the FHA at 24 hours shows that the capability of the AP1000 design to meet the regulatory dose acceptance criteria is not sensitive to the decay time. The staff's position is that in order to not include a technical specification LCO for the decay time, the design basis FHA dose analysis must assume a decay time that is clearly less than the time physically needed to begin moving fuel assemblies out of the core following unit shutdown for refueling. The staff does not consider 100 hours to be short enough. The applicant did not revise the design basis FHA dose analysis, which continues to include the 100-hour decay time assumption. Additionally, in DCD Tier 2 Section 15.7.4.5 discussing of the evaluation of the FHA at 24 hours, the applicant did not discuss the impact of the decay time on control room habitability due to an FHA at 24 hours. The staff does not consider this issue to be resolved. This is Open Item 15.3.7-1.

15.3.8 Offsite Radiological Consequences of Liquid Tank Failure

SRP Section 15.7.3, "Postulated Radioactive Releases due to Liquid-Containing Tank Failures," contains guidance on review of the failure of a liquid containing tank. The acceptance criteria specified in this SRP section are based on meeting the following regulations:

- General Design Criterion GDC 60 as it relates to the radioactive waste management system being designed to control release of radioactive materials to the environment.

- 10 CFR Part 20 as it relates to radioactivity in effluents to unrestricted areas.

The failure of the most limiting (i.e., in terms of offsite radiological consequences) liquid radwaste system (WLS) equipment outside the containment does not result in radionuclide concentrations in water at the nearest potable water supply in an unrestricted area exceeding the liquid effluent concentration limits for the corresponding radionuclides specified in Appendix B to 10 CFR Part 20 (Table 2, Column 2). Specific design features to mitigate the effects of failure are incorporated in the design of the WLS, if it does not meet the above requirements of 10 CFR Part 20.

In the AP1000 design, tanks containing radioactive fluids are located inside plant structures. In the event of a tank failure, the liquid would be drained by the floor drains to the auxiliary building sump. From the sump, the water would be directed to the waste holdup tank. Because SRP Section 15.7.3 states that credit cannot be taken for liquid retention by unlined building foundations, the assumption is made that release to the environment is possible.

DCD Section 15.7.3 includes a commitment for a COL action item to perform a site-specific offsite radiological consequence analysis, including the corresponding source term resulting from a postulated liquid tank failure. The staff finds this commitment to be acceptable because the assessment of offsite radiological consequences of liquid tank failures depends upon site-specific parameters, such as the mode of transport of radioactive fluid resulting from the failure to the region of potable water supply, the location of potable water supply, the characteristics of the soil through which the transport occurs, and the available dilution by water-bodies before the radioactive liquid reaches the potable water supply. The staff will evaluate the site-specific analysis in accordance with SRP Section 15.3.7 for each COL applicant referencing the AP1000 standard design. This is COL action item 15.3.8-1.

15.3.9 Radiological Consequences of Loss of Spent Fuel Pool Cooling

For the radiological consequences analysis of each design basis accident, Westinghouse additionally evaluated the added radiological consequences of spent fuel pool (SFP) boiling due to loss of SFP cooling capability.

The SFS is designed to perform the following functions:

- remove heat from the SFP and IRWST
- remove radioactive corrosion and fission products from the SFP, IRWST and refueling cavity
- transfer water between the IRWST and the refueling cavity for refueling operations

The system consists of redundant trains. Each train includes a pump, a heat exchanger, a filter, and a demineralizer. However, the SFP cooling system is a non-safety-related system. Therefore, the applicant assumed, and the staff agrees, that a loss of SFP cooling capability should be analyzed coincident with DBAs.

The loss of SFP cooling could result in the pool reaching boiling and a portion of the radioactive iodine in the SFP water could be released to the environment. Without actions to provide makeup water to the SFP, boiling is assumed to commence at 8.8 hours after loss of SFP cooling capability. The applicant has calculated that the dose consequences from this source is less than 0.1 mSv (0.01 rem) TEDE both offsite and to the control room operators. This dose is added to the dose consequences of each DBA to find the overall dose consequences of the DBA coincident with loss of SFP cooling.

To verify the applicant assessment, the staff performed an independent radiological consequence calculation for the loss of the spent fuel pool cooling capability. The major parameters and assumptions used by the staff are provided in Table 15.3-11 of this report, and

the results of the staff's radiological consequence analyses are provided in Table 15.3-1 of this report. The offsite radiological consequences calculated by the staff are consistent with those calculated by the applicant.

15.3.10 Conclusions

The staff has reviewed the radiological consequences analyses of the design basis accidents described DCD Tier 2 Chapter 15 for the AP1000 design. Based on the evaluation discussed above, the staff concluded that the AP1000 design meets the acceptable offsite dose criteria for these accidents, with the exception of the LOCA, as discussed in Section 15.3.6 of this report.

As discussed above, the staff has not completed its review of the applicant's control room habitability radiological consequences analyses. The staff has not yet made a determination as to whether the control room habitability dose criterion in GDC-19 is met for the AP1000 design.

The staff identified one COL Action Item and four open items as follows.

- COL Action Item 15.3.8-1 regarding a site-specific analysis of the offsite radiological consequences of a liquid tank failure:

  The COL applicant should perform a site-specific offsite radiological consequences analysis of the postulated liquid tank failure to confirm that the plant meets the applicable regulations on radioactive waste management systems and radiological effluents. The COL applicant should submit in the plant specific application the radiological consequences analysis for the staff to review and approve.

- Open Item 15.3-1 regarding the use of the AP600 aerosol removal coefficients for the AP1000 design:

  The staff has not completed its evaluation of the use of the AP600 aerosol removal coefficients. The staff is performing independent analyses to confirm information provided by the applicant. The staff may have additional clarifying questions, but does not require any additional information at this time.

- Open Item 15.3-2 regarding the control room habitability radiological consequences analyses:

  The staff has not completed its evaluation of the hypothetical reference control room atmospheric dispersion factors, as discussed in Section 2.3.4 of this report. When this evaluation is completed, the staff will complete its review of the design basis accident control room habitability radiological consequences analyses.

- Open Item 15.3.6-1 regarding the LOCA radiological consequences analysis:

  The staff has not completed its evaluation of the LOCA radiological consequences analysis because of Open Items 15.3-1 and 15.3-2 discussed above. When these Open

Items are resolved, the staff will complete its review of the design basis LOCA offsite and control room radiological consequences analyses.

- Open Item 15.3.7-1 regarding the FHA radiological consequences analyses:

  The applicant has not provided sufficient analysis of the radiological consequences of the FHA taking into account the absence of a TS LCO for decay time before movement of fuel. The applicant should provide an analysis of the radiological consequences of an FHA that occurs 24 hours post-shutdown, including both the offsite and control room doses.

Table 15.3-1
Radiological Consequences of Design-Basis Accidents
(Total Effective Dose Equivalent)

Postulated Accident	EAB	LPZ	Control Room
Loss of coolant accident	TBD*	TBD*	TBD*
Main steam line break outside containment:			
With accident initiated iodine spike	6 mSv (0.6 rem)	15 mSv (1.5 rem)	TBD*
With preaccident iodine spike	2 mSv (0.2 rem)	4 mSv (0.4 rem)	TBD*
Reactor coolant pump shaft seizure	5 mSv (0.5 rem)	1 mSv (0.1 rem)	TBD*
Rod ejection accident	18 mSv (1.8 rem)	14 mSv (1.4 rem)	TBD*
Fuel handling accident	22 mSv (2.2 rem)	5 mSv (0.5 rem)	TBD*
Small line break accident	12 mSv (1.2 rem)	3 mSv (0.3 rem)	TBD*
Steam generator tube rupture			
With accident initiated iodine spike	6 mSv (0.6 rem)	4 mSv (0.4 rem)	TBD*
With preaccident iodine spike preaccident	12 mSv (1.2 rem)	4 mSv (0.4 rem)	TBD*
Spent fuel pool boiling	n/c**	<0.1 mSv (<0.01 rem)	TBD*

* TBD, to be determined - see Open Items 15.3-1 and 15.3.6-1 for the LOCA, 15.3-2 for control room doses
** n/c, not calculated

Table 15.3-2
Assumptions Used To Evaluate the Radiological Consequences of the
Main Steam Line Break Accident Outside Containment

Parameter	Value
Power level, MWt	3468
Reactor primary coolant iodine concentrations	
Accident initiated iodine spike, µCi/gm DEI-131	1.0
Preaccident iodine spike, µCi/gm DEI-131	60
Steam generator in faulted loop	
Initial water mass, lb	3.03E+5
Primary to secondary leak rate, gpd	500
Iodine partition coefficient	1
Steam generator in intact loop	
Primary to secondary leak rate, gpd	500
Iodine partition coefficient	0.01
Steam released, lb	
0 to 2 hr	3.0335E+5
2 to 8 hr	1.225E+04
Ratio of iodine release rate from fuel during iodine spike to that during steady-state operation	500
Reactor primary coolant mass, lb	3.84E+5
Duration of accident, hr	72
Atmospheric dispersion values, sec/m^3	
EAB	
0 to 2 hours	6.0E-4
LPZ	
0 to 8 hours	1.35E-4
8 to 24 hours	1.00E-4
1 to 4 days	5.40E-5
Control room analysis parameters	Table 15.3.9

Table 15.3-3
Assumptions Used to Evaluate the Radiological Consequences of the
Reactor Coolant Pump Shaft Seizure Accident (Locked Rotor)

Parameter	Value
Power level, MWt	3468
Fraction of fuel rods failed	0.16
Fraction of core activity in failed fuel rod gap	
I-131	0.08
Kr-85	0.10
Other iodines and noble gases	0.05
Alkali metals	0.12
Reactor primary coolant iodine concentrations	
Preaccident iodine spike, µCi/gm DEI-131	60
Secondary coolant mass, lb	6.06E+5
Primary to secondary leak rate, lb/hr	350
Iodine and alkali partition coefficient	0.01
Steam released, lb	
0 to 1.5 hr	6.48E+5
Leak flashing fraction	
0 to 60 min	0.04
> 60 min	0
Reactor primary coolant mass, lb	3.7E+5
Duration of accident, hrs	1.5
Atmospheric dispersion values, sec/m^3	
0 to 2 hours, EAB	6.0E-4
0 to 8 hours, LPZ	1.35E-4
Control room analysis parameters	Table 15.3.9

Table 15.3-4 (Sheet 1 of 2)
Assumptions Used To Evaluate the Radiological Consequences
of the Rod Ejection Accident

Parameter	Value
Power level, MWt	3468
Peaking factor	1.65
Fraction of fuel rods failed	0.1
Fraction of fission-product inventory released to coolant from perforated fuel rods	
Iodines and noble gases	0.1
Alkali metals	0.12
Fraction of fuel rods melted	0.0025
Fraction of fission-product inventory released to coolant from melted fuel rods	
Iodines and alkali metals	0.5
Noble gases	1.0
Initial reactor coolant iodine activity, µCi/gm DEI-131	60
Reactor coolant mass, lb	3.7E+5
Duration of accident, days	30
Iodine chemical form fractions	
Organic	0.0015
Elemental	0.0485
Particulate	0.95
Secondary system release path	
Primary to secondary leak, lb/hr	350
Leak flashing fraction	0.04
Secondary coolant mass, lb	6.06E+5
Duration of steam release from secondary system, sec	1800
Steam released from secondary system, lb	1.08E+5
Partition coefficient in steam generators	
Iodine	0.01
Alkali metals	0.001

Table 15.3-4 (Sheet 2 of 2)
Assumptions Used To Evaluate the Radiological Consequences
of the Control Rod Ejection Accident

Parameter	Value
Containment leakage release path	
Containment leak rate, % per day	0.10
Airborne activity removal coefficients, hr^{-1}	
Elemental iodine	1.7
Organic iodine	0
Particulate iodine or alkali metals	0.1
Decontamination factor limit for elemental iodine removal	200
Time to reach decontamination factor limit, hr	3.1
Atmospheric dispersion values, sec/m^3	
EAB	
0 to 2 hours	6.0E-4
LPZ	
0 to 8 hours	1.35E-4
8 to 24 hours	1.00E-4
1 to 4 days	5.40E-5
4 to 30 days	2.20E-5
Control room analysis parameters	Table 15.3.9

Table 15.3-5
Assumptions Used To Evaluate the Radiological Consequences
of the Small Line Break Outside Containment Accident

Parameter	Value
Power level, MWt	3468
Reactor primary coolant iodine concentrations	
Accident initiated iodine spike, µCi/gm DEI-131	1.0
Preaccident iodine spike, µCi/gm DEI-131	60
Ratio of iodine release rate from fuel during iodine spike to that during steady-state operation	500
Reactor coolant mass, lbs	3.7E+5
Duration of accident, minutes	30
Sample line break flow, gpm	130
Fraction of reactor coolant flashing	0.41
Atmospheric dispersion values, sec/m^3	
0 to 2 hours, EAB	6.0E-4
0 to 8 hours, LPZ	1.35E-4
Control room analysis parameters	Table 15.3.9

Table 15.3-6
Assumptions Used To Evaluate the Radiological Consequences
of the Steam Generator Tube Rupture Accident

Parameter	Value
Power level, MWt	3468
Reactor primary coolant iodine concentrations	
Accident initiated iodine spike, µCi/gm DEI-131	1.0
Preaccident iodine spike, µCi/gm DEI-131	60
Steam generator in faulted loop	
Initial water mass, lb	3.03E+5
Primary to secondary leak rate, gpd	500
Iodine partition coefficient	1
Steam generator in intact loop	
Primary to secondary leak rate, gpd	500
Iodine partition coefficient	0.01
Steam released, lb	
0 to 2 hr	3.64E+5
2 to 8 hr	7.15E+5
Ratio of iodine release rate from fuel during iodine spike to that during steady-state operation	335
Reactor primary coolant mass, gm	1.63E+5
Duration of accident, hr	8
Atmospheric dispersion values, sec/m^3	
EAB	
0 to 2 hours	6.0E-4
LPZ	
0 to 8 hours	1.35E-4
8 to 24 hours	1.00E-4
1 to 4 days	5.40E-5
4 to 30 days	2.20E-5
Control room analysis parameters	Table 15.3.9

Table 15.3-7
Assumptions Used to Evaluate the Radiological Consequences
of the Loss-of-Coolant Accident

Parameter		Value
Power level, MWt		3468
Core activity released to the containment atmosphere, fraction		
Nuclide Group	Gap Release (0 - 0.5 hr)	In-vessel Release (0.5 - 1.3 hr)
Noble gases	0.05	0.95
Iodines	0.05	0.35
Alkali metals	0.05	0.25
Tellurium group		0.05
Strontium and barium		0.02
Noble metals		0.0025
Cerium group		0.0005
Lanthanide group		0.0002
Iodine chemical form fractions		
Organic		0.0015
Elemental		0.0485
Particulate		0.95
Primary containment leakage, weight percent/day		0.1
Primary containment free volume, ft^3		2.06E+6
Elemental iodine deposition removal coefficient, hr^{-1}		1.7
Decontamination factor limit for elemental iodine removal		200
Removal coefficients for particulates		See Table 15.3-8
Accident duration, days		30
Atmospheric dispersion values, sec/m^3		
EAB		
0 to 2 hours		6.0E-4
LPZ		
0 to 8 hours		1.35E-4
8 to 24 hours		1.00E-4
1 to 4 days		5.40E-5
4 to 30 days		2.20E-5
Control room analysis parameters		Table 15.3.9

Table 15.3-8
Aerosol Removal Rates Used to Evaluate
Loss-of-Coolant Accident

Time Post-Release (hours)	Removal Rates ($hour^{-1}$)
0.0 to 0.5	to be determined*
0.5 to 1.8	to be determined
1.8 to 3.8	to be determined
3.8 to 13.8	to be determined
13.8 to 24	to be determined

* See Open Items 15.3-1 and 15.3.6-1

Table 15.3-9
Assumptions Used to Evaluate the Radiological Consequences
to Control Room Operators Following a Design Basis Accident

Parameter	Value
Accident release modeling	
Main Steam Line Break	Table 15.3-2
Locked Rotor Accident	Table 15.3-3
Rod Ejection Accident	Table 15.3-4
Small Line Break Outside Containment	Table 15.3-5
Steam Generator Tube Rupture	Table 15.3-6
LOCA	Table 15.3-7
Fuel Handling Accident	Table 15.3-10
Control room free volume, ft^3	3.57E+4
Initial interval prior to actuation of emergency habitability system	
Air intake flow, cfm	1925
Intake filter efficiencies	N/A
Interval with operation of emergency habitability system	
Activity level at which emergency habitability system is actuated, Ci/m^3 of dose equivalent I-131	2.0E-6
Actuation of emergency habitability system for LOCA, hr	0.2622
Flow from compressed air bottles, cfm	60
Unfiltered inleakage, cfm	5
Bottled air depletion time, hr	72
Interval after depletion of bottled air supply	
Air intake flow, cfm	1700
Intake filter efficiencies	N/A
Recirculation flow	N/A
Breathing rate of operators in control room for the course of the accident, m^3/sec	3.47E-4
Atmospheric dispersion values	Table 15.3-9a
Control room operator occupancy factors	
0 to 24 hr	1
24 to 96 hr	0.6
96 to 720 hr	0.4

Table 15.3-9a
Atmospheric Dispersion Factors (χ/Q)
For Control Room Habitability Accident Dose Analysis
(<u>not</u> found acceptable by NRC staff pending resolution of DSER Open Item 2.3.4-1)

χ/Q (sec/m^3) at Control Room HVAC Intake for Identified Release Points

	Elevated Containment Release	Ground Level Containment Release	Secondary Side Release Points	Fuel Handling Area	Fuel Building Relief Panel
0 - 2 hrs	1.2E-3	1.2E-3	2.0E-2	2.0E-3	3.0E-3
2 - 8 hrs	8.0E-4	6.3E-4	1.8E-2	1.5E-3	2.0E-3
8 - 24 hrs	4.0E-4	3.0E-4	8.0E-3	8.0E-4	1.0E-3
24 - 96 hrs	4.0E-4	3.0E-4	7.0E-3	8.0E-4	1.0E-3
96 -720 hrs	3.0E-4	2.6E-4	6.0E-3	7.0E-4	9.0E-4

χ/Q (sec/m^3) at Control Room Door for Identified Release Points

	Elevated Containment Release	Ground Level Containment Release	Secondary Side Release Points	Fuel Handling Area	Fuel Building Relief Panel
0 - 2 hrs	4.0E-4	6.6E-4	2.5E-3	1.0E-3	1.0E-3
2 - 8 hrs	2.0E-4	3.8E-4	2.0E-3	6.0E-4	6.0E-4
8 - 24 hrs	1.0E-4	1.9E-4	1.0E-3	3.0E-4	3.0E-4
24 - 96 hrs	9.0E-5	1.8E-4	9.0E-4	3.0E-4	3.0E-4
96 -720 hrs	8.0E-5	1.6E-4	8.0E-4	2.5E-4	2.5E-4

Table 15.3-10
Assumptions Used To Evaluate the Radiological Consequences
of a Fuel Handling Accident

Parameter	Value
Power level, MWt	3468
Peaking factor	1.65
Number of fuel assemblies in core	157
Number of assemblies damaged	1
Reactor shutdown time before fuel movement, hr	100
Core fractions released from damaged rods	
I-131	0.08
Other iodines	0.05
Kr-85	0.10
Other noble gases	0.05
Iodine chemical form fractions	
Organic	0.0015
Elemental	0.0485
Particulate	0.95
Iodine effective pool decontamination factor	200
Duration of accident, hr	2
Atmospheric dispersion values, sec/m^3	
0 to 2 hours, EAB	6.0E-4
0 to 8 hours, LPZ	1.35E-4
Control room analysis parameters	Table 15.3.9

Table 15.3-11
Assumptions Used To Evaluate the
Radiological Consequences of Spent Fuel Pool Boiling

Parameter	Value
Initial activity in spent fuel pool, Ci	
I-131	3.18
Other nuclides	None modeled
Fuel stored in spent fuel pool from 10 years of operation	
Includes 68 assemblies from a recent refueling	
Amount of I-131 diffusing into pool over 30 day period, Ci	1.94
Initial pool water temperature, °F	120
Time to initiate pool boiling, hr	8.8
Steaming rate, lb/hr	
8.8 - 24 hr	16,200
24 - 48 hr	16,000
48 - 72 hr	15,700
72 - 88 hr	15,420
88 - 120 hr	15,250
120 - 168 hr	14,930
≥ 168 hr	14,500
Iodine partition coefficient	0.01
Atmospheric dispersion values, sec/m^3	
LPZ	
8 - 24 hr	1.00E-4
1 - 4 days	5.40E-5
4 - 30 days	2.20E-5
Offsite breathing rate, m^3/sec	
8 - 24 hr	1.8E-4
> 24 hr	2.3E-4
Control room analysis parameters	Table 15.3.9