# 15. TRANSIENT AND ACCIDENT ANALYSES

## 15.1 <u>Introduction</u>

In the AP1000 Design Control Document (DCD) Tier 2, Chapter 15, "Accident Analyses," the applicant discussed the analysis of various design-basis transients and accidents. The applicant used 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, "Reactor Design," for fuel design limits, GDC 15, "Reactor Coolant System Design," for the reactor coolant pressure boundary (RCPB) pressure limits, and the requirements of Title 10, Section 50.46, of the <u>Code of Federal Regulations</u> (10 CFR 50.46) for the performance of the emergency core cooling system (ECCS).

The staff of the U.S. Nuclear Regulatory Commission (NRC) has reviewed the AP1000 transient and accident analyses in the DCD, in accordance with Chapter 15, "Accident Analysis," of NUREG-0800, which defines the agency's Standard Review Plan (SRP).

### 15.1.1 Event Categorization

The applicant assigned 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 (ATWSs)

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 this category is broader than the category of increase in heat removal from the primary system in the SRP and RG 1.70 and reflects the AP1000 design, it is acceptable.

The applicant also grouped 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 Condition 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 their effects on the consequences of Conditions II, III, and IV events should be considered. Condition II events may occur during a calendar year for a particular plant. Condition III events 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, "General Design Criteria for Nuclear Power Plants," 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-frequency 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 in the DCD is consistent with the NRC's licensing approach.

In DCD Tier 2, Section 15.0.1, the applicant listed the design-basis events analyzed under Conditions II, III, and IV. These events are generally consistent with the current licensing practice. However, the DCD lists the complete loss of forced reactor coolant flow event and the single rod cluster control assembly (RCCA) withdrawal event at full-power conditions as Condition III—infrequent faults. The event categorization is inconsistent with SRP Sections 15.3.1 and 15.4.3, which classify the complete loss of RCS flow and the withdrawal of an RCCA as Condition II—moderate-frequency events, with the acceptance criterion that specifies no violation against the safety limit of the departure from nucleate boiling ratio (DNBR). Nonetheless, the applicant analyzed the complete loss of forced reactor coolant flow event, as presented in DCD Tier 2, 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.

Per WCAP-9272-A, "Westinghouse Reload Safety Evaluation Methodology," Westinghouse was previously allowed to classify the withdrawal of an RCCA event as a Condition III event for reactors that it manufactured because of the event's very low probability of occurrence. Because the manufacturer of the existing pressurized-water reactors (PWRs) also designed the AP1000 reactor, the applicant's classification of the withdrawal of an 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 can be used to mitigate the events. In Westinghouse letter ET-NRC-93-3804, dated January 22, 1993, the applicant responded to the staff's request for additional information (RAI) 440.31 for the AP600 review. The applicant 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 nonconsequential random, independent failure must occur in order to disable the system

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

For the AP1000 design, Westinghouse applied, as indicated in DCD Tier 2, Chapter 15, the same approach as in the AP600 design, indicated above in the specified situations for non-safety-related systems or components to be used for the analyses of the design-basis events.

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

For Case (b), the applicant assumed 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 an RCCA is analyzed from an at-power condition. Before the initiating fault causing the RCCA withdrawal, the MFCS should operate and maintain steam generator (SG) inventory. If a failure exists in the MFCS, alarms in the control room or abnormal control system performance should detect it before the start of the RCCA withdrawal event. The staff concludes that the assumption of MFCS continued operation is acceptable because 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 timeframe of the initiating event is extremely low.

For Case (c), as discussed in the response to RAI 440.061 and summarized in DCD Tier 2, Table 15.0-8, the applicant credited 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 (LONF), 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 (including the turbine stop, control valves, turbine bypass valves, moisture separator reheat steam supply control valve, and main steam branch isolation valves in the analysis of inadvertent opening of SG safety valves, steamline break (SLB), and SGTR events)

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 that a TS be established for a structure, system, or component (SSC) that is assumed to function or actuate in a design-basis analysis for the mitigation of specified events. In its response to RAI 440.061, the applicant indicated that it complied 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 AP1000 TS Table 3.3.2-1, “Engineering Safeguards Actuation System Instrumentation,” include applicable modes, surveillance requirements (SRs), and trip setpoints for the main feedwater pump trip and pressurizer heater trip, respectively. Section 3.7.2 of the AP1000 TS provides the LCOs for the main steam branch isolation valves and the MSIV backup valves. 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 for 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 SRs and LCOs in the TSs to ensure the reliability of the following systems or components:

  — feedwater pump trip breakers and redundant pressurizer heater trip breakers
  — MSIV backup valves and 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, the applicant performed the Chapter 15 analysis assuming a loss of offsite 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, "Electric Power Systems," of 10 CFR Part 50, Appendix A, which requires the analysis of AOOs and postulated accidents 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 assumed that a LOOP and the resulting coastdown of the reactor coolant pumps (RCPs) occurs 3 seconds after the turbine trip. DCD Tier 2, Section 8.2, provides the basis for the 3-second delay. 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, allows the generator to provide voltage support to the grid and maintain adequate voltage to the RCPs for significantly longer than the assumed 3 seconds.

- The COL applicant interface item in 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) 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, Item 8.3, (that the protective devices controlling the switchyard breakers are set with consideration for preserving the plant grid connection following a turbine trip) is especially important in generator output circuit breaker designs to ensure that the opening of the switchyard breakers following a turbine trip does not interrupt the backfeed offsite circuit through the generator main stepup transformer.

- This design does not use automatic transfers of RCP buses, which 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 undervoltage or exciter overcurrent.

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 (discussed in Section 8.2.3.4 of this report). The staff has also reviewed the DCD Tier 2, Chapter 15, analysis and found that the applicant considered LOOP in all of the applicable analyzed events and applied 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, the methods used for transient and accident analyses include the following computer codes:

- TWINKLE—This multidimensional spatial neutronics code uses an implicit finite-difference method to solve the two-group transient neutronics 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-NP-A (nonproprietary), "TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code," issued January 1975, the NRC has approved this code 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 (nonproprietary), “VIPRE-01 Modeling and Qualification for Pressurized Water Reactor Non-LOCA Thermal-Hydraulic Safety Analysis,” issued October 1999, the NRC has approved this code for the core thermal-hydraulic (T-H) 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 T-H calculations is acceptable.

- COAST—As documented in CENPD-98-A, “Coast - Code Description,” issued April 1973, the NRC has approved this code for use in calculating the reactor coolant flow coast transient for any combination of active and inactive RCPs and forward and reverse flow in the hot- or cold-legs. The NRC approved the code for the PWRs manufactured by the former Combustion Engineering (CE) (now merged as part of Westinghouse). The COAST code uses equations of conservation of momentum formulated for each of the flowpaths of the COAST model, assuming unsteady-state, one-dimensional 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 square of the flow velocity. 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 RCP coastdown transient.

- FACTRAN—As documented in WCAP-7908-A (nonproprietary), “FACTRAN - A FORTRAN-IV code for Thermal Transients in a UO2 Fuel Rod,” issued June 1972, 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-NP-A (nonproprietary), “LOFTRAN Code Description,” issued April 1984, the NRC previously approved this code to allow Westinghouse to analyze system responses to non-LOCA events for conventional Westinghouse PWRs. LOFTRAN simulates a multiloop system using a model containing a reactor vessel (RV), hot- and cold-leg piping, SGs, and a pressurizer. The code also includes a point kinetics model, including reactivity effects of the moderator, fuel, boron, and control rods. The secondary side of the SG 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. WCAP-14234, "LOFTRAN and LOFTTR2 AP600 Code Applicability Document," issued June 1997, describes the LOFTRAN modifications. WCAP-14307, "AP600 LOFTRAN-AP and LOFTTR2-AP Final Verification and Validation Report," issued August 1997, documents the test data comparisons that support the LOFTRAN modifications.

- LOFTTR2—As documented in WCAP-10698-P-A (proprietary) and WCAP-10750-NP-A (nonproprietary), "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," issued August 1985, 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 breakflow model, a two-region SG 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. WCAP-14234 documents these changes.

- 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 (proprietary), "NOTRUMP—A Nodal Transient Small Break and General Network Code," issued August 1985, and WCAP-10054-P-A (proprietary), "Westinghouse Small-Break ECCS Evaluation Model Using the NOTRUMP Code," issued August 1985, the NRC previously approved the NOTRUMP code for the SBLOCA analysis. WCAP-14807, Revision 5, "NOTRUMP Final Validation Report for AP600," issued August 1998, documents the modified version of the NOTRUMP code for the AP600 application.

- WCOBRA/TRAC-LBLOCA—As documented in WCAP-12945-P, Revision 2, "Code Qualification Document for Best Estimate LOCA Analysis," Volumes 1 through 5, issued March 1998, the NRC has approved this best estimate (BE) code in a safety evaluation dated June 28, 1996, for large-break loss-of-coolant accident (LBLOCA) analysis. WCAP-14171, Revision 2, "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident," issued March 1998, documents the modified version of the WCOBRA/TRAC code for the AP600 application. Westinghouse letter NSD-NRC-97-5171 dated June 10, 1997, documents its auxiliary code, HOTSPOT, which is updated for the AP600.

- WCOBRA/TRAC-LTC and WGOTHIC—WCOBRA/TRAC is also used for the post-LOCA long-term cooling (LTC) analyses. WCAP-14776, Revision 4, "WCOBRA/TRAC, OSU Long-Term Cooling Final Validation Report," issued March 1998, documents the code verification for the LTC analyses. The WGOTHIC code, documented in WCAP-14407, Revision 3, "WGOTHIC Application to AP600," issued April 1998, is used to calculate containment boundary conditions for LBLOCA and post-LOCA LTC. The staff previously reviewed and accepted the application of WCOBRA/TRAC and WGOTHIC for LTC calculations, as discussed in Chapter 21 of this report.

In support of the AP1000 application, the applicant submitted WCAP-15644-P, "AP1000 Code Applicability Report," issued March 2004, for the staff to review. WCAP-15644-P documents the applicant's 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 design. The safety analysis codes include LOFTRAN, LOFTTR2, NOTRUMP, WCOBRA/TRAC, and WGOTHIC. The staff reviewed the report and concluded in a March 25, 2002, letter from J.E. Lyons (NRC) 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:

- Table 2 of the enclosure to the March 25, 2002, letter listed the transients and accidents that Westinghouse proposed to analyze with the LOFTRAN code. The staff limited its review of LOFTRAN usage by Westinghouse to this set. The use of the code for other analytical purposes will require additional justification.

- In the preapplication review, the 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. Chapter 21 of this report includes a review of this material.

In addressing the staff's review question regarding compliance with the limitations imposed by the staff on the use of the LOFTRAN and LOFTTR2 codes, the applicant provided its response to RAI 440.054 and indicated 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. Because 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 provided the assessment addressing the applicability of NOTRUMP, WCOBRA/TRAC, and WGOTHIC to the safety analysis for the AP1000 design. The staff has reviewed the applicant's compliance assessment and documented its evaluation in Section 21.6.2 of this report for NOTRUMP, Sections 21.6.3 and 21.6.4 of this report for WCOBRA/TRAC, and Section 21.6.5 of this report for WGOTHIC.

### 15.1.5 Steam Generator Middeck Plate Induced Level Measurement Uncertainty

Westinghouse has issued three Nuclear Service Advisory Letters (NSALs), NSAL-02-3 and Revision 1, NSAL-02-4, and NSAL-02-5, which document the concerns with the Westinghouse-designed SG water-level setpoint uncertainties. NSAL-02-3 and its revision, dated February 15 and April 8, 2002, respectively, deal with the uncertainties in the SG water-level measurement caused by the placement of the middeck plate between the upper and lower taps. These uncertainties affect the low-low level trip setpoint (used in the analysis for events such as the feedwater line break (FLB), ATWS, and SLB). NSAL-02-4, dated February 19, 2002, deals with the uncertainties in the measurement created because the calculation does not reflect the void content of the two-phase mixture above the middeck plate. These uncertainties affect the high-high level trip setpoint. NSAL-02-5, dated February 19, 2002, deals with potential inaccuracies in the initial conditions assumed in safety analyses affected by SG water level. The safety analyses may not be bounding because the velocity head under some conditions may increase the uncertainties in the SG water-level control system. The staff requested, via RAI 440.062, the applicant to discuss (1) the AP1000 design that accounts for all the 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's response to RAI 440.062 stated that measurement uncertainties for the reactor protective system and engineered safety feature (ESF) actuation system instrumentation 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," issued April 1996. The methodology can be used for performing setpoint studies independent of the hardware used for the protection system, and, therefore applies to the AP1000 design. The setpoint study will include applicable uncertainties discussed in the referenced NSALs. Using the methodology in WCAP-14605, plant nominal setpoints are calculated by adding the channel allowance from the setpoint study to the setpoints 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 middeck-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 item was designated as draft safety evaluation report (DSER) Open Item 15.1.5-1. In addressing DSER Open Item 15.1.5-1, Westinghouse provided its response in a letter from M.M. Corletti (Westinghouse) to the NRC dated July 1, 2003, "Transmittal of Westinghouse Responses to Open Items Identified in the AP1000 Draft Safety Evaluation Report." In the response, Westinghouse indicated that Item E of Revision 1 to RAI 440.002 addressed the COL information in DCD Tier 2, Section 7.1.6, related to instrumentation setpoint uncertainty calculations by the COL applicant upon selection of the installed plant instrumentation. In accordance with the current practice, the safety analyses will become plant specific when the COL applicant performs the setpoint study to develop plant-specific TS setpoints based on the safety analysis values, adding instrumentation uncertainty, as well as uncertainties related to the effects of the SG middeck plate on SG-level measurement. The TS setpoints are changed to reflect instrumentation uncertainties. Westinghouse stated, and the staff agreed, that the plant safety analyses are not changed because the revised TS setpoints are based on the safety analyses documented in the DCD. The COL actions to establish appropriate plant-specific setpoints, as discussed in DCD Tier 2, Section 7.1.6, add the appropriate plant-specific values to the TS setpoints used in the safety analyses. Therefore, the staff concludes that no other COL action is needed to reperform any safety analyses, and DSER Open Item 15.1.5-1 is resolved. This is COL Action 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. Section 15.3 of this report presents the staff's evaluation of the analysis for radiological releases.

### 15.2.1 Increase in Heat Removal from the Primary System

In DCD Tier 2, Section 15.1, the applicant presented 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 an SG relief or safety valve, (5) SLB, and (6) inadvertent operation of the PRHR HX. The staff provides its evaluation of the analytical results in the following sections.

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

Failure of a low- or high-pressure heater train may cause a decrease in feedwater temperature, a moderate-frequency event. A reduction in feedwater temperature decreases reactor coolant temperature, which, in turn, causes an increase in core power because of the effects of the negative moderator coefficient of reactivity. Because the rate of energy change is reduced as load and feedwater flows decrease, the transient initiated from 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 the 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 °C (79.5 °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), which results in a power increase of 12 percent. Section 15.2.1.3 of this report discusses the staff's review of the event with an excessive increase in secondary steam flow.

#### 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. The SG high-2 water-level signal trip prevents the continuous addition of excessive feedwater by closing 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 calculates the nuclear power transient, the RCS flow coastdown, and the primary pressure and temperature transient. FACTRAN calculates the heat flux based on the nuclear power and flow from

LOFTRAN. VIPRE-01 calculates the departure from DNBR during the transient, using the heat flux from FACTRAN and the flow from LOFTRAN.

The applicant analyzed both the no-load case and the full-power case. For the no-load condition, the applicant assumed a feedwater control valve malfunction resulting in a step increase to 120 percent of nominal feedwater flow to one SG. The applicant assumed a feedwater temperature at the low value of 4.4 °C (40 °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 did not assume a LOOP in the no-load case. The results of the analysis show 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 stemming from the effects of the negative moderator coefficient of reactivity. Section 15.2.4.1 of this report provides the staff's review and approval of the analysis of an uncontrolled RCCA bank withdrawal event.

The applicant's analysis for the limiting case is based on initial full-power conditions with an 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. An SG high-2 level trip signal actuates a reactor trip and an associated turbine trip. In addressing the issue of a LOOP, the applicant assumed 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 a 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 an 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 indicated in its response to RAI 440.063 that an 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 applicant performed the analysis using an 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 falls below 110 percent of the RCS design pressure, and the calculated DNBRs for the transient remain above the safety limit DNBR defined in DCD Tier 2, Section 4.4. As a result, the analysis satisfies the acceptable criteria defined in SRP Section 15.1.2. 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)

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

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

- 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
- 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 of full power. Each case is analyzed without taking credit for pressurizer heaters. At the initiation of the event, the RCS pressure and temperature are assumed at their full-power values for the DNBR calculation. The safety DNBR limit, as described in WCAP-11397-P-A, "Revised Thermal Design Procedure," issued April 1989, includes uncertainties in initial conditions. In DCD Tier 2, Sections 15.1.4 and 15.1.5 analyze steam flow increases greater than 10 percent, and Sections 15.2.1.4 and 15.2.1.5 of this report evaluate them.

In demonstrating the capability of the plant for the cases with automatic rod control, the applicant took no credit for delta T 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 it determined that no single active failure in these systems or equipment will adversely affect the consequences of the event. In considering the effects of a LOOP, the applicant assumed a reactor trip with a coincident turbine trip followed by a LOOP 3 seconds later. The LOOP primarily causes the RCPs 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 the RCPs lose power. Therefore, the applicant's analysis indicates that the minimum DNBRs predicted during the event will occur before 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. Because 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 <u>Inadvertent Opening of an SG Relief or Safety Valve (DCD Tier 2, Section 15.1.4)</u>

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

In assessing the effects of the negative MTC, the applicant's analysis assumes the most negative MTC corresponding to the end-of-life rodded core with the most reactive RCCA in its fully withdrawn position. Availability of offsite power is assumed to maximize the cooldown effect. Because 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. Because the turbine is initially in the trip condition for the plant at 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. It identified the limiting single failure as a failure of one CMT discharge valve to open. The applicant also made the following assumptions to maximize the cooldown effects:

- A typical capacity steam flow rate of 236 kilograms per second (kg/s) at 8.2 MegaPascals (MPa) (520 pounds mass per second (lbm/s) at 1200 pounds per square inch absolute (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 RCPs are assumed to be operating initially.

- No moisture is assumed in the blowdown steam.

- Manual actuation of the PRHR system is assumed at the initiation of the event.

The applicant used the LOFTRAN code to analyze the event. During the transient, the low cold-leg temperature “S” signal automatically actuates the CMT injection and the associated tripping of the RCPs. Boron solution at 3400 parts per million (ppm) enters the RCS, 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 the departure from nucleate boiling (DNB) does not occur, thereby satisfying the acceptance criteria in SRP Section 15.1.4. Therefore, the staff concludes that the analysis is acceptable.

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

An 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 MTC, the RCS temperature decrease results in an addition of positive reactivity, which increases the core power level. The SG pressure decrease initiates a reactor trip when low pressure in the steam system produces a safeguards “S” signal. The “S” signal initiates the actuation of the CMTs, which, 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. Ultimately, the borated water from the CMTs shuts down the reactor.

The applicant used the LOFTRAN code to calculate the system transient and the VIPRE-01 code to determine whether DNB had occurred for the core transient conditions calculated by the LOFTRAN code. The applicant analyzed a double-ended rupture at no-load conditions with no decay heat as the limiting case. Because the SGs have integral flow restrictors with a 0.13 square meter $(m)^2$ (1.4 square feet $(ft)^2$) 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; therefore, the analysis assumes this limiting break area.

Because the average coolant temperature for a core tripped from at-power conditions is higher than at no-load conditions, and energy is stored in the fuel, the RCS for a core tripped from at-power conditions contains more stored energy than at no-load conditions. 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 assumed 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 the failure of the MSIV in the intact SG to close, the applicant took 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 made 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 is assumed for the end-of-life rodded core with the most reactive RCCA stuck out.

- 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 RCPs are assumed to be operating initially.

- 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.

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

- The 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, actuation of the CMTs will 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 RCP operation reduces the rate of boron injection into the core, which increases the potential for the core to return to criticality after reactor trip.

- The plant protection system automatically provides a safety-related signal that initiates the coastdown of the RCPs coincident with CMT actuation. Because 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 MSIVs fully close in less than 10 seconds from receipt of a closure signal.

In response to RAI 440.067, which addresses the staff's concern regarding the effect of the timing of a LOOP on the analysis of the limiting SLB case, the applicant analyzed two full-break SLB cases initiated with the reactor at no-load conditions, one with offsite power available throughout the event, and one with offsite power loss simultaneous with the SLB at the start of the event. The SLB analysis shows that for the case with LOOP, the RCPs begin coasting down at the initiation of the transient, and, for the case with offsite power 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 affect the return to power. The calculated peak core heat flux for the case with offsite power available was slightly greater than that for the LOOP case (3.17 percent versus 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 the event 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 temperature remains below saturated temperatures, confirming that the SLB analysis falls within the applicable range of the LOFTRAN code (also discussed in Section 15.1.4 of this report).

- The values used for input parameters, resulting in a maximum cooldown effect and the greatest potential for 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 Section 15.1.5 for an SLB analysis.

15.2.1.6 Inadvertent Operation of the PRHR (DCD Tier 2, Section 15.1.6)

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

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

For this analysis, the applicant used LOFTRAN for the system response calculation, FACTRAN for the heat flux determination, and VIPRE-01 for the DNBR calculation and assumed a negative moderator coefficient for the end-of-life rodded core. The applicant generated the core properties used in the LOFTRAN code for reactivity feedback calculations 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 only when their operation results in more severe conditions. The analysis considered cases both with and without automatic rod control. The reactor trips on high neutron flux, and the analysis does not credit overtemperature and overpower delta T trips.

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 it determined that no single active failure in these systems or equipment will adversely affect the consequences of the event.

In considering the effects of a LOOP, the applicant assumed 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 RCS flow coastdown begins. The resulting power reduction compensates for the reduced flow encountered once the RCPs lose power. The applicant's analysis indicates that the minimum DNBRs predicted during the event occur before flow coastdown begins. With the assumption of no reactor trip occurring 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 that the assumptions used in the analysis are conservative for the reasons stated above and, therefore, are 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 resulting in decreasing steam flow, (2) loss of external electrical load, (3) turbine trip, (4) inadvertent closure of MSIVs, (5) loss of condenser vacuum and other events resulting in turbine trip, (6) loss of ac power to the station auxiliaries, (7) LONF flow, and (8) feedwater system pipe break. The staff has reviewed the applicant's analyses, as discussed in Sections 15.2.2.1 through 15.2.2.8 of this report.

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

The AP1000 design includes no SG pressure regulators whose failure will cause a decreasing steam flow transient. Therefore, this event does not apply to the AP1000 design.

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

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

The turbine trip event bounds this loss of external electrical load 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 from a slower termination of steam flow will result in a lower peak RCS pressure. Section 15.2.2.3 below discusses the staff's evaluation of the turbine trip analyses.

15.2.2.3 Turbine Trip (DCD Tier 2, Section 15.2.3)

Signals, including generator trip, low condenser vacuum, loss of lubricating oil, turbine thrust bearing failure, turbine overspeed, manual trip, and reactor trip, may initiate the turbine trip event. 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 performed the analysis for this event using 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, RCS pressure, and RCS temperature are assumed to be at their nominal values consistent with steady-state, full-power operation. The DNBR limit includes uncertainties in initial conditions as described in WCAP-11397-P-A. 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, the analysis takes no credit 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 availability of the pressurizer safety valves is assumed to reduce the pressure increase during the transient. In considering the effects of a LOOP, the applicant has assumed that offsite power will last for 3 seconds after the turbine trip. The applicant also has considered plant systems and equipment, as discussed in DCD Tier 2, Section 15.0.8, that are available to mitigate the effects of the event and it 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 considered both minimum and maximum reactivity feedback cases. The applicant also considered the event with and without credit for the effect of pressurizer spray in reducing the reactor coolant pressure. The applicant analyzed each case 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 available offsite power and takes no credit for the effect of pressurizer spray in reducing the RCS pressure.

The staff finds that (1) the analysis used computer codes previously approved by the NRC and adequate assumptions to maximize the peak pressure, (2) the calculated peak RCS pressure for the limiting turbine trip case falls below 110 percent of the RCS design pressure, (3) pressurizer overfilling does not occur, and (4) the calculated minimum DNBR is within the safety DNBR limit, thus satisfying the acceptance criteria of SRP Section 15.2.3. 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 of this report.

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. Because 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 the loss of the condenser vacuum. Therefore, the analytical results reviewed and discussed in Section 15.2.2.3 of this report 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)

A complete loss of the offsite grid, accompanied by a turbine-generator trip, may cause the loss of ac power, a moderate-frequency event. In terms of the removal of decay heat, this event is more severe than the turbine trip event because, for this event, an RCS flow coastdown accompanies the decrease in heat removal by the secondary system, 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 used the LOFTRAN code to perform the RCS system response analysis following a plant LOOP. The analysis credits only safety-related systems 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 5.1-1979, “Decay Heat Power in Light-Water Reactors,” decay heat data 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 narrow-range SG level. The assumption of a LOOP coincident with the reactor trip is more conservative than 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 HX heat transfer coefficients are assumed to be at low values associated with the low flow rate caused by the RCP trip. In RAI 440.074, the staff requested the applicant to 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, Revision 3, “AP600 Passive Heat Removal Heat Exchanger Test Final Report,” issued April 1997, which the NRC previously reviewed and approved, in a letter dated March 25, 2002, from J.E. Lyons (NRC) to W.E. Cummins (Westinghouse), for application to the AP1000 as part of the AP1000 preapplication review. In WCAP-12980, the results of calculations using the Dittus-Boelter correlation show that the predicted values reasonably agree with the PRHR test data. The operating conditions for the PRHR tests include 1.1 Lpm to 37.8 Lpm (0.3 gpm to 10.0 gpm) for the single HX tube flow, 121 °C to 343 °C (250 °F to 650 °F) for the primary temperature, and 0.446 MPa to 16.0 MPa (50 psig to 2300 psig) for the primary water pressure. The applicant calculated the PRHR tube flow and primary temperature during the loss of ac power event and showed that the flows and temperatures fall within the range of the PRHR test conditions. During the loss of ac power transient, the primary pressure increased from 15.5 MPa to 17.2 MPa (2250 psia to 2500 psia), which slightly exceeds the test range.

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

The applicant used the LOFTRAN, FACTRAN, and VIPRE-01 codes with the revised thermal design procedure (RTDP) described in WCAP-11397-P-A 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. The analysis includes uncertainties in initial conditions, as described in the RTDP, in determining the DNBR limit during the transient. The functionality of the SG safety valves and pressurizer safety valves is assumed for steam releases, and the CMTs are assumed to actuate when the PRHR HX cools down the RCS enough to initiate a low cold-leg temperature "S" signal.

In considering the effects of a LOOP, the applicant assumed 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 a LOOP initiates at the beginning of the event. Section 15.2.3.2 of this report discusses the results of the complete loss of RCS flow 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 it determined that the worst single active failure is the 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. The loss of ac power is assumed following the reactor trip. Loss of ac power causes the RCPs to coast down. The PRHR HX actuates on a low narrow-range SG water level coincident with low startup feedwater flow rate, and the CMTs actuate when the PRHR HX cools the RCS enough to initiate 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 remove 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)

An LONF flow event, a moderate-frequency event, may be caused by feedwater pump failures, valve malfunctions, or loss of ac power sources. Following an event involving an LONF, 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. Either a low narrow-range SG water-level signal, coincident with a low startup feedwater flow rate signal, or a low wide-range SG water-level signal actuates the PRHR HX. 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 the 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 long-tem decay heat removal.

The applicant performed 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. A low wide-range SG water level actuates the PRHR HX.

In considering the effects of measurement uncertainties, the initial temperature and pressurizer pressure are assumed to be 4 °C and 0.446 MPa (7 °F and 50 psi) below the nominal values. In response to RAI 440.075, related to the staff’s concern about the adequacy of the initial temperature and pressure assumed in the analysis, the applicant replied that during the LONF event, the availability of ac power is assumed 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, which is calculated with the LOFTRAN code as a function of the RCS pressure and, 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, discussed in DCD Tier 2, Section 15.0.8, that are available to mitigate the effects of the event, and it 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 assumed that the power loss and the resulting coastdown of the RCPs occur 3 seconds after the turbine trip for DNBR calculations. The LOOP causes a coastdown of the RCPs. The applicant showed that the scenario of the loss of ac power event, for which the RCPs trip instantaneously, bounds the LONF transient event followed by the consequential LOOP after turbine trip. The analysis of the loss of ac power event, presented in DCD Tier 2, Section 15.2.6, shows that the calculated minimum DNBR exceeds the safety DNBR limits. Therefore, the minimum DNBR for the loss of feedwater event also exceeds the safety DNBR limits.

The applicant performed the analyses for the peak RCS pressure and LTC for the loss of feedwater event using approved methods. 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. For LTC, 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 remove the decay heat. Thus, the SRP acceptance criteria for the LONF 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)

An 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 SGs. 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. 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 may actuate a reactor trip. During the event, either a low narrow-range SG water level, coincident with a low startup feedwater rate signal, or a low wide-range SG water-level signal may actuate the PRHR HX. In the long term, the PRHR HX removes the decay heat and provides continuous core heat removal capability. A low cold-leg temperature “S” signal may actuate the CMTs.

The applicant assumed a break in a feedwater line between the check valve and the SG with a double-ended rupture of the largest feedwater line. The applicant assumed a double-ended break area of 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. In a followup to RAI 440.076, the staff requested the applicant to discuss the analysis used to determine the limiting break case for the AP1000 design. 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 the DCD with FLB analysis for break sizes of 100 percent, 50 percent, 25 percent, and 10 percent of the feedwater nozzle, assuming the break occurred at the initial transient time. The results confirmed that the DCD case is the limiting case resulting in the highest RCS water inventory in the pressurizer and the highest peak primary pressure.

The applicant performed the analysis of this event using the LOFTRAN computer code. The initial power is assumed at 102 percent of the rated power. Initiation of the reactor trip is assumed when the low narrow-range SG level setpoint 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 occurring simultaneously with the pipe break, with the LOOP occurring during the FLB accident, and without a LOOP, and identified (in the response to RAI 440.077) that the FLB with the LOOP occurring at the time of the break is the limiting case, resulting in the highest RCS pressure. In the analysis for the limiting FLB case, the low wide-range SG water level is assumed to actuate the PRHR HX, with a maximum delay time of 15 seconds to initiate automatic alignment of the PRHR HX valves. In addition, the applicant took 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 ESFs required to function include the PRHR, CMTs, and steam isolation valves.

The applicant 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 it determined that the worst single active failure is the 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 assumed 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. In addition, the analysis assumes a low pressurizer safety-valve setpoint. Both assumptions will result in a lower peak RCS pressure and, thus, are not conservative. The AP600 FLB analysis made the same nonconservative assumptions. 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 it had reanalyzed the event without pressurizer spray operable and with the pressurizer safety-valve setpoint at its normal value. The confirmatory analysis showed a peak RCS pressure of 18.08 MPa (2624 psia) and an increase of 27.6 kPa (4 psi) as compared to the DCD case, and it confirmed that the effects of the nonconservative assumptions on the calculated peak RCS pressure are small. Because the applicant's FLB analysis documented in the DCD shows that the RCS pressure response during an FLB event for the AP1000 design is similar to that of the AP600 design, the effects of the nonconservatism in modeling the pressurizer spray and pressurizer safety valves for the AP1000 FLB analysis will also be small. In addition, the AP1000 FLB analysis shows that the calculated peak RCS pressure is less than 17.93 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 nonconservative assumptions of the pressurizer spray and pressurizer safety-valve models discussed above.

The applicant performed the FLB analysis using 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 LTC, 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 cause a coastdown of the RCPs. The applicant showed that, for the transient up to the reactor trip and complete insertion of control rods (where the minimum DNBR occurs), the scenario of the loss of ac power event, for which the RCPs trip instantaneously, bounds the FLB event followed by the consequential LOOP after turbine trip. The analysis of the loss of ac power event, presented in DCD Tier 2, Section 15.2.6, shows that the calculated minimum DNBR exceeds the safety DNBR limits. Therefore, the minimum DNBR for the FLB event also exceeds the safety DNBR limits.

Because the applicant used NRC-approved methods, the assumptions used in the analysis are adequate in maximizing RCS pressure and minimizing the calculated DNBRs, and the results of the analysis meet the acceptance criteria of SRP Section 15.2.8 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) RCP shaft seizure (locked rotor), and (4) RCP shaft break. The applicant has also identified the limiting case with regard to the ability of the RCS boundary and fuel rod cladding to withstand the consequences of transients. The staff has reviewed the applicant's analysis, as discussed in Sections 15.2.3.1 through 15.2.3.4 of this report.

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

A mechanical or electrical failure in an RCP, or a fault in the power supply to the pumps supplied by an RCP bus, may cause partial loss of RCS flow, a moderate-frequency event. The low primary coolant flow reactor trip signal in any reactor coolant loop provides protection against this event.

The applicant analyzed the partial loss of flow event using 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. 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 RTDP described in WCAP-11397-P-A. In the DNBR calculations, the initial reactor power, pressure, and RCS 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 used the least negative MTC and a large absolute value of the Doppler power coefficient. The applicant calculated the RCP flow coastdown based on RCS pressure losses and RCP characteristics. Reactor coolant fluid momentum is neglected to obtain a low coastdown flow, which will result in lower calculated DNBRs.

In considering the effects of a LOOP, the applicant assumed 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. This delay to turbine trip is a feature of the AP1000 reactor trip system (RTS). The LOOP primarily causes the remaining operating RCPs to coast down. The analysis shows that the LOOP will have no effect on the calculated minimum DNBR, because 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, discussed in DCD Tier 2, Section 15.0.8, that are available to mitigate the effects of the event, and it determined that no single active failure in these systems or equipment adversely affects the consequences of the events.

Because an event involving the loss of three of the four RCPs is not credible, the applicant did not analyze the consequences of this event. In addition, the core flow would be much lower for an event involving the 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. In DCD Tier 2, Sections 15.3.1 and 15.3.2 analyze and discuss the loss of two RCPs and the loss of four RCPs, respectively.

The applicant analyzed the event using 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 simultaneous loss of electrical power to all RCPs may cause a complete loss of forced flow from the 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 the RCP underspeed trip actuates a reactor trip following the loss of power supply to all pumps at power. Section 15.2.3.1 of this report discusses the methods and assumptions used in the analysis. 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 it 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)

An instantaneous seizure of an RCP rotor may cause RCP shaft seizure, and an instantaneous failure of an RCP shaft may cause an RCP shaft break. 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 convective heat transfer coefficient, and the reactor coolant in the tube side cools down while the shell-side temperature increases because of steam flow through the turbine reducing to zero upon plant trip. The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the SGs, causes 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, because the slightly faster RCP flow coastdown for the shaft seizure event results in a lower minimum DNBR.

The applicant analyzed the more limiting RCP shaft seizure 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. The analysis takes no credit for restoration of offsite power before initiation of shutdown cooling.

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 it determined that no single active failure in these systems or equipment adversely affects the consequences of the events. The applicant analyzed this event using the LOFTRAN code for the system response and the FACTRAN code for the heat flux calculation at the hot spot. The NRC has approved both codes for these analyses. The low reactor coolant flow signal actuates the reactor trip. The analysis takes no credit 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 indicated, in the response to RAI 440.080, that the calculated minimum DNBR is above the safety-limit DNBR and thus assures no rod failure. However, for the purpose of calculating dose releases, the applicant conservatively assumed that 16 percent of rods are damaged. The results show that the dose release limits are met even with the assumed 16 percent of fuel rods damaged. Section 15.3 of this report discusses the staff's evaluation of the radiological calculations.

The applicant used NRC-approved methods with results that show that 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 presented the analytical results of events caused by 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) malfunction or failure of the flow controller in a boiling water recirculation loop that results in an increased reactor coolant flow rate, (6) CVCS 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 following sections discuss the staff's evaluation of the analytical results.

#### 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)

A malfunction of the reactor control or rod control systems may cause an uncontrolled RCCA bank withdrawal from a subcritical or low-power startup condition. 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 provide protection against this event.

For the analysis of this transient, the applicant used 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 low value of Doppler-power coefficient and the least negative 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 exceeds 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/min (45 in./min). The DNBR calculation assumes the most limiting axial and radial power shapes associated with the two highest-worth banks in their high-worth position. 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 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 it 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 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 worth and reactivity coefficients used in the analysis and found that they maximize the heat flux, thereby minimizing 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, "Protection System Functions," in that the reactivity control system can be initiated automatically so that specified acceptable fuel design limits are not exceeded. In addition, the applicant meets GDC 25, "Protection System Requirements for Reactivity Control Malfunctions," 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)

A malfunction of the reactor control or rod control systems may cause an uncontrolled withdrawal of an RCCA bank in the power operating range, a moderate-frequency event. Such an event causes an increase in fuel and coolant temperature as a result of the core-turbine power mismatch. Reactor trips, including the high neutron flux trip, overpower and overtemperature delta T trips, and pressurizer high-pressure and pressurizer water-level trips, provide plant protection.

The applicant performed the analyses using 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. 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 WCAP-11397-P-A. In the DNBR calculations, the initial reactor power, pressure, and RCS 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 exceed 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 include instrumentation and setpoint uncertainties, and the delays for trip actuation are assumed to be at the maximum values. The applicant analyzed cases with both minimum and maximum reactivity coefficients and performed 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 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 it 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 assumed 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 Section 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. Asymmetric power distributions sensed by in-core or ex-core neutron detectors or core exit thermocouples, rod deviation alarms, or rod position indicators can detect misaligned rods. The deviation alarm alerts the operator to rod deviation from the group position in excess of 5 percent of span.

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

The following sections discuss 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.

#### 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 (PLS) 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 determined that the automatic operating mode bounds the manual operating mode and is the limiting DNBR case.

The applicant analyzed the rod drop events in the automatic control mode using the nuclear models with the computer codes described in DCD Tier 2, Table 4.1-2, for the calculation of the hot channel factor, the LOFTRAN code for the system response, and the VIPRE-01 code for the DNBR calculation. The results show that the calculated minimum DNBR exceeds 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 satisfies 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 RCCA drop event is acceptable.

15.2.4.3.2 Analysis for a Misaligned Full-Length Assembly

For RCCA misalignment situations, the applicant analyzed 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 using the approved methods described in DCD Tier 2, Table 4.1-2. Uncertainties in initial conditions, as described in WCAP-11397-P-A, are included in determining the DNBR limit during the transient. The analysis shows that the minimum DNBR exceeds the safety DNBR limit. Therefore, the staff concludes that the analysis is acceptable because 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, the applicant classified the single assembly withdrawal as an infrequent event for the AP1000 design. The event categorization is consistent with that approved by the staff for 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 using the approved methods described in DCD Tier 2, Table 4.1-2. Uncertainties in initial conditions, as described in WCAP-11397-P-A, are included in determining the DNBR limit during the transient. In response to RAI 440.081, the applicant indicated that less than 4 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 for one RCCA which is fully withdrawn. For the purpose of calculating dose releases, the applicant conservatively assumed that 5 percent of the fuel rods failed. The assumption of fuel failure for the dose release calculation is more conservative than the guidance in SRP Section 4.4, which states that all rods that experience DNB (in this case, less than 4 percent of the fuel rods) 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 meets the requirements of GDC 27, "Combined Reactivity Control Systems Capability," 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 percent) of the fuel rods may experience cladding perforation. The dose release calculation results show that the release acceptance criteria are met. Therefore, the staff concludes that the analysis is acceptable. Section 15.3 of this report discusses the staff's evaluation of the radiological consequence calculations.

#### 15.2.4.4 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature (DCD Tier 2, Section 15.4.4)

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

Because the TSs (described in DCD Tier 2, Chapter 16, TS 3.4.4) do not allow operation with an RCP inoperable for Modes 1 and 2, the applicant did 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 does not apply 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)

Failures of the demineralized water transfer and storage system (DWS) or CVCS because of control system or operator error or mechanical failure cause an inadvertent boron dilution. 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 boric acid and blended flow rates, the status of the CVCS makeup pumps, and the boron dilution rate deviation alarm indicate the dilution rate. 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 available alarms and indications to alert the operator to a boron dilution event include (1) a high flux at shutdown alarm, (2) indicated source-range neutron flux count rates, (3) an audible source-range neutron flux rate, and (4) a 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 applicant performed the analysis using the method consistent with that employed 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 flowpath to represent CVCS fluid transportation.

All cases discussed below assume a dilution flow rate of 12.6 liters per second (L/sec) (200 gallons per minute (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. The boric acid tank (BAT), which contains borated water, supplies makeup water during refueling.

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

In Modes 3, 4, and 5, the analysis assumed a shutdown margin of 1.6 percent delta K/K (the minimum value required by the AP1000 TSs for the shutdown modes) and the minimum initial reactor coolant volumes. Following the AP1000 TS LCO 3.4.8 requirements, the applicant assumed the operation of one RCP. In maximizing the effect of the boron dilution, the applicant used 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 surgeline and the RV upper head volume. For Mode 4, the water volume assumed in the analysis is the water volume of the RV without the RV upper head volume when the normal residual heat removal system (RNS) 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 midloop operations. The source-range flux multiplication signal is assumed to actuate an alarm in the control room and close the DWS isolation valves when the neutron flux increases by 60 percent over any 50-minute period (per Item 15.a of AP1000 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 the start of the 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 isolation of the DWS valves terminates the boron dilution and maintains the plant in a subcritical condition. The staff determined that the analysis meets the guidance in SRP Section 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, rod control is manual. The applicant performed an analysis of an 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 requirements of TSs 3.1.1 and 3.4.4, the applicant assumed an available shutdown margin of

1.6 percent delta K/K and operation of four RCPs. The initial RCS water assumed in the analysis is the water volume included in all the RCS volumes except the pressurizer and surgeline. Calculation of the SG tube volume accounts for 10-percent tube plugging. 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 Section 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 analyzed both the manual mode and the automatic mode cases. Both cases use the same initial RCS water volume as for Mode 2 discussed above. For the manual mode case, the analytical results show that a reactor trip on the overtemperature delta T will initiate closure of the DWS isolation valves and terminate the boron dilution without the occurrence of a posttrip return to criticality. Because 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, the slow insertion of the control rods to avoid the reactor trip compensates for an increase in the power and temperature caused by a boron dilution event. Because a reactor and turbine trip does not occur as predicted in the analysis for the case in automatic mode, the consequential LOOP 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 pretrip 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 an available time interval from a low-low rod insertion limit alarm, attributable to boron dilution, to loss of shutdown margin of about 328 minutes (DCD Tier 2, Table 15.4-1). The staff finds that the applicant has demonstrated compliance with the guidance in SRP Section 15.4.6, in that the redundant pretrip alarms should alert the operator to the 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.

The analysis shows that (1) for Mode 6, the design and procedures prevent the inadvertent boron dilution events, (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 can alert an operator to a boron dilution event, and the operator has sufficient time (328 min) 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 core subcriticality and, therefore, concludes that it is acceptable.

In support of the boron mixing model used in the analysis, the applicant specified a required minimum core flow rate. Specifically, TS LCO 3.4.8 requires that at least one RCP operate 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 to support the conclusion that the required core flow rate is sufficient to provide the well-mixed flow condition assumed in the boron dilution analysis. The applicant replied that the process of selecting 630 L/sec (10,000 gpm) included general consideration of the results reported in NUREG/CR-2733, “Experimental Data Report for LOFT Boron Dilution Experiment L6-6,” issued June 1982. As discussed in EGG-LOFT-5867, “Quick-Look Report on LOFT Boron Dilution Experiment L6-6,” issued 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 four-loop Trojan PWR. With its four cold-legs, the general configuration of the AP1000 inlet plenum region is similar to that of a four-loop plant.

The LOFT 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, intended to ensure adequate boron mixing in current Westinghouse-designed plants, allow operation in the applicable mode with a single operating RHR pump. 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, documents the results of the tests. 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, though 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 LOFTs have been used to support the typical plant TSs that generally accept an RHR flow in the vicinity of 189 L/sec (3000 gpm) as sufficient to justify the perfect mixing assumption modeled in the boron dilution analysis.

For the AP1000, the minimum core flow required by the TS exceeds the flow rates considered in the LOFT and currently accepted as providing adequate mixing in the operating plants. In addition, 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 operate at a minimum of 25-percent rated speed, which produces a flow rate of 1,239 L/sec (19,688 gpm). This SR indicates that the total RCP flow is almost twice the required 630 L/sec (10,000 gpm) core flow and much greater than the 189 L/sec (3,000 gpm) value that is typically applied to operating plants. Since the general configuration of the AP1000 inlet flow plenum region is similar to that of a four-loop Westinghouse plant, the LOFTs that were scaled to a Westinghouse plant and are used to support the boron mixing model for the current Westinghouse plants apply to the AP1000 for selection of the minimum core flow rate to assure a well-mixed flow condition. In addition, 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 (3,000 gpm) that is typically applied to the operating plants and supported by the LOFT results. Therefore, the staff concludes that the required minimum core flow gives reasonable assurance that it is sufficient to provide the well-mixed flow conditions considered in boron dilution events that were analyzed to address the guidance in SRP Section 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 indicated 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 performed an analysis of the consequences of a loading error.

The applicant used the NRC-approved methods documented in WCAP-10965-P-A, "ANC: Westinghouse Advanced Nodal Computer Code," issued 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 provided 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 fuel assembly, is loaded near the core periphery.

The analysis described above shows that resulting power distribution effects will either be detected by the startup test involving the in-core detector system (and hence be 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, "Instrumentation and Control," 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 fall within the acceptable measurement uncertainty, ensuring no fuel failure and satisfying the SRP Section 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 include 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 it determined that no single active failure in these systems or equipment adversely affect the consequences of the events. The staff has reviewed this analysis in accordance with SRP Section 15.4.8.

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

The applicant analyzed two sets of cases for the rod ejection event, one initiated at hot full-power (HFP) and one initiated at hot zero-power (HZP). The analysis of both of these cases uses both beginning-of-cycle (BOC) and end-of-cycle (EOC) kinetics. DCD Tier 2, Table 15.4-3, lists the values of the initial plant parameters (power level, ejected rod worth, delayed neutron fraction, and trip reactivity) assumed in the analysis. The analysis credits the high neutron flux trip (high and low setting) to trip the reactor. The results show that the calculated values of hot spot radially averaged fuel enthalpy 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 fall below the safety limit of 280 cal/g specified in SRP Section 15.4.8, “Spectrum of Rod Ejection Accidents.” The calculated values also fall 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 Section 15.4.8 with respect to the RCS pressure limit.

In considering the effects of a LOOP, the applicant assumed that the power loss and the resulting RCP coastdown occur 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 rod 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 experience DNB as a result of the rod ejection event. For the purpose of calculating dose releases, the applicant conservatively assume that 10 percent of the fuel fails. 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 Section 15.4.8. However, the staff, the industry, and the international community agree that burnup degradation in the margin to low-enthalpy fuel failure is likely to be regained by application of more detailed three-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 fall below 100 cal/gm (R.O. Meyer, R.K. McCardell, H.M. Chung, D.J. Diamond, and H.H. Scott, “A Regulatory Assessment of Test Data for Reactivity-Initiated Accidents,” Nuclear Safety, Volume 37, Number 4, October–December 1996, pages 271–288). In addition, a generic analysis performed by the applicant that assumes low-enthalpy fuel failure shows that the radiological consequences of rod ejection accidents meet the acceptance criteria specified in Appendix A to SRP Section 15.4.8. As indicated in the response to RAI 440.181, the applicant’s generic analysis is predicated on conservative treatment of the experimental fuel data applied to existing and planned cores operating within approved burnup limits for PWRs. Therefore, the staff concludes that, although the SRP Section 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 Section 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. Section 15.3 of this report includes the review of the radiological releases.

The applicant performed the analysis using acceptable methods. The results of the analysis show that the calculated values of peak fuel enthalpy fall below the acceptable limit specified in SRP Section 15.4.8, the calculated peak RCS pressure does not exceed the RCS emergency limits (Service Level C), and the radiological consequences meet the SRP Section 15.4.8 acceptance criteria. The staff finds that the analysis meets the acceptance criteria of SRP Section 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)

Operator action, a false electrical actuation signal, or a valve malfunction can cause spurious CMT operations. The DCD presents the results of the most limiting case, a CMT inadvertently actuated by operator error or a mechanical failure resulting in the opening of two valves in the CMT discharge lines. During the event, the high-3 pressurizer water-level signal actuates to trip the reactor, followed by the PRHR actuation and eventually by an “S” signal, which then actuates the second CMT. The applicant analyzed the case using the LOFTRAN code and established the following initial conditions to maximize pressurizer water level:

- 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 $^\circ$C (7 $^\circ$F) below nominal.

- The pressurizer spray system and automatic rod control are operable.

- A least-negative MTC, 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. The pressure drop of the CMT injection and balance lines is minimized to maximize the CMT flow injected into the primary system. In response to RAI 440.085, the applicant indicated that modeling high pressure drops through the PRHR loop minimizes the PRHR heat transfer capability. A higher pressure drop limits the PRHR flow and reduces the calculated value of the primary-side heat transfer coefficient. In addition, a maximum TS value for PRHR tube plugging and a 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, discussed in DCD Tier 2, Section 15.0.8, that are available to mitigate the effects of the event, and it 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 assumed that a power loss and the resulting coastdown of the RCPs occur 3 seconds after the turbine trip.

The analysis assumes that an inadvertent opening of the CMT discharge valves, which results in the one CMT injecting borated water, initiates the event. 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 a 3-second delay, a consequential LOOP is assumed, and the RCPs are tripped. The cold-leg temperature increases, resulting in an increased CMT injection rate, because of the increased driving head from the density decrease in the pressure 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. Initiation of the PRHR, with appropriate delay time, is then assumed. The primary and secondary pressures increase initially because of the assumed unavailability of the non-safety-related control systems, but eventually decrease 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 from 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 used the LOFTRAN code for the analysis and adequately identified the limiting case. 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 that increases RCS inventory. Operator action, an electrical actuation signal, or a valve failure may cause the CVCS to malfunction. The DCD presents the results of the most limiting case, the CVCS malfunction caused by an operator error, resulting in the startup of two CVCS pumps to deliver the flow to the RCS. The applicant analyzed CVCS malfunction cases using the LOFTRAN code and established the following initial conditions to maximize the pressurizer water level:

- The reactor power is at 102 percent of nominal, the pressure is 344.7 kPa (50 psi) above nominal, and the RCS temperature is at 3.6 °C (6.5 °F) above nominal.

- The pressurizer spray system is operable.
- A least-negative MTC, 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 case that models explicit operator actions after the reactor trip.

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 it 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 assumed that a power loss and resulting coastdown of the RCPs occur 3 seconds after the turbine trip.

The analysis assumes that a CVCS malfunction that results in injection from two CVCS pumps initiates the event. As the CVCS injection flow increases RCS inventory, the pressurizer water volume begins increasing while the primary system cools 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 a 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 open, 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 used the LOFTRAN code for the analysis with adequate inputs and appropriately identified 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 limit 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 provided an analysis of events that may decrease the RCS inventory. These events include (1) an inadvertent opening of a pressurizer safety valve or inadvertent operation of the automatic depressurization system (ADS), (2) a break in an instrument line or other lines from the reactor coolant boundary that penetrate the containment, (3) an SG tube failure, and (4) a LOCA resulting from a spectrum of postulated piping breaks within the RCPB. The following sections discuss the applicant’s analysis and the staff’s evaluation.

### 15.2.6.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the Automatic Depressurization System (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 reactivity 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 considered an opening of both first stage ADS flowpaths to be the limiting case, because operation of these valves results in a greater depressurization rate than ADS Stages 2 and 3 valves given 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 ADS Stage 1 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, discussed in DCD Tier 2, Section 15.0.8, that are available to mitigate the effects of the event, and it determined that no single active failure in these systems or equipment adversely affected the consequences of the event. In addressing a LOOP, the applicant assumed that a power loss and resulting coastdown of the RCPs occur 3 seconds after the turbine trip. The analysis shows that a LOOP has no effect on the calculated minimum DNBR, because 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 the minimum DNBR occurs before initiation of a LOOP.

To perform the analysis of these events, the applicant used 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, RCS 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 analyzed 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 DNBR 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 released directly 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 sample line orifices limit the maximum breakflow to 8.2 L/s (130 gpm).

Both the isolation valves inside and outside containment are open only during sampling, and the loss of sample flow will provide an indication of the break to plant operators. A break in a sample line releases radioactivity which will actuate area and air radiation monitors. Because the operator sees multiple indications, the applicant assumed that, 30 minutes after initiation of a break, the operator would isolate the sample line and terminate further release of primary fluid discharged to the 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 fall within the 10 CFR 50.34(a)(1)(ii)(D)(*1*) limits. Thus, the staff determines that the analysis meets the SRP Section 15.6.2 acceptance criteria. Therefore, the staff concludes that the analysis is acceptable. Section 15.3 of this report discusses the staff's evaluation of the radiological release calculations.

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. The failure of an SG U-tube may cause this accident.

The analysis for the SGTR event consists of two parts, SG overfill calculation and the calculation of the SG mass releases used to evaluate the radiological consequence.

The applicant performed an analysis with LOFTTR2 to demonstrate that the AP1000 design features can prevent the SG from overfilling with water. To maximize the SG water increase, the applicant identified the limiting single failure as 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 include a 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 demonstrate that the AP1000 protection system and passive system design features will prevent the SG from overfilling with water during an SGTR.

For the SG mass release calculation, the applicant performed the SGTR analysis using the LOFTTR2 code for a case with complete severance of a single SG tube. At the initiation of an SGTR, the reactor is assumed to be 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 tripping the RCPs (resulting from a LOOP) has been determined to maximize flashing of primary-to-secondary break flow, consequently maximizing radiological releases. The LOOP is assumed to trip the reactor. 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 the pressurizer heater addition are assumed at the initiation of the event (even though 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. Because 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, discussed in DCD Tier 2, Section 15.0.8, that are available to mitigate the effects of the event, and it identified that the most limiting single failure is a failed-open PORV on the affected SG. The applicant assumed that the single failure occurs coincidently with the low-2 pressurizer-level signal, maximizing the integrated RCS-to-secondary break flow. The automatic closure of the associated block valve on a low steamline pressure protection system signal isolates the SG PORV.

The analysis shows that after the reactor trip, low pressurizer pressure generates a safeguard “S” signal. The “S” signal results in CMT actuation and PRHR system actuation. Opening 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 breakflow 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 breakflow terminates, and the system reaches a stable condition. The analysis shows that the PRHR can remove the core decay heat and prevent the unaffected PORV from opening. During the transient, the CMTs remain full, 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 used the LOFTTR2 computer code together with conservative and acceptable assumptions to maximize the primary-to-secondary leakage. 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 the PRHR and CMTs can achieve LTC, 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. Section 15.3 of this report discusses the staff's evaluation of the radiological release.

15.2.6.4 <u>Spectrum of Boiling-Water Reactor Steam System Piping Failure Outside Containment (DCD Tier 2, Section 15.6.4)</u>

This section of the DCD does not apply to the AP1000 design, which is a PWR design.

15.2.6.5 <u>Loss-of-Coolant Accident (DCD Tier 2, Section 15.6.5)</u>

In DCD Tier 2, Section 15.6.5, Westinghouse presents the LOCA analysis results. The applicant's analyses examine SBLOCAs, LBLOCAs, and post-LOCA LTC.

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

- The calculated peak cladding temperature (PCT) is less than 1204 °C (2200 °F).
- The calculated total oxidation of the cladding is within 0.17 times the total cladding thickness before oxidation.
- 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, reacts.
- Any calculated changes in core geometry will be such that the core remains amenable to cooling.
- 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 AP1000 is designed to keep the reactor core cooled and covered with water by means of passive safety systems which do not require the start and operation of pumps to provide makeup water. No operator action is required to actuate and control the passive protective systems. Active systems using pumps are also available for activation and control by the operator in the event of an SBLOCA. However, the design-basis analysis of the AP1000 design does not credit the operation of these active systems.

During an 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 RV downcomer by way of the direct vessel injection (DVI) line. Water will also begin to circulate through the PRHR HX 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, which will retain water in the lower elevations of the reactor system and around the core and will 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 begin to sequence open when the CMT water volume drops to 67.5 percent. The ADS Stage 4 valve (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, and 3 causes the reactor pressure to decrease. When the pressure reaches approximately 4.83 MPa (700 psig), the accumulator tanks, which contain borated water pressurized with nitrogen, will inject into the RV by way of the 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 RV.

In DCD Tier 2, Section 15.6.5.4B.2, “Small-Break LOCA Analysis Methodology,” the applicant described the three elements of the AP1000 SBLOCA evaluation model as the NOTRUMP computer code, the NOTRUMP homogeneous sensitivity model, and the CHF assessment during accumulator injection. As described in the following discussion, the applicant applied the three elements in the analyses for the AP1000 design. The results demonstrate that the design of the AP1000 adequately mitigates the consequences of postulated design-basis SBLOCA events. Section 21.6.2 of this report discusses the staff’s evaluation and acceptance of the NOTRUMP code and the SBLOCA evaluation model for the AP1000.

The applicant performed SBLOCA analyses using the NOTRUMP computer code for the period immediately after the break occurs until IRWST flow is fully established to the reactor core. After this time, the reactor is considered to be in LTC. Section 15.2.7 of this report discusses the applicant’s analyses for LTC. The NRC staff conducted its evaluation of SBLOCA for the AP1000 design in accordance with SRP Section 15.6.5 to ensure that the acceptance criteria listed in 10 CFR 50.46 will not be exceeded.

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. 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 NOTRUMP analyses were made conservative by assuming decay heat at 120 percent of the ANS 5.1-1971 Standard, as required by Appendix K to 10 CFR Part 50. 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 design. For most of the NOTRUMP analyses, the containment was assumed to remain at atmospheric pressure. The use of atmospheric pressure maximizes pressure losses out of the three ADS-4 valves assumed to remain operable, which delays the time when sustained inventory injection to the RV from the IRWST can occur. DCD Tier 2, Section 15.0.8, discusses the plant systems and equipment that are assumed to be available to mitigate the effects of the event.

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

- the inadvertent opening of both 10.16-cm (4-in.) ADS-1 valves (atmospheric containment pressure)
- a cold-leg break of 5.08 cm (2 in.) equivalent diameter in the loop without the pressurizer (atmospheric containment pressure)
- the double-ended rupture of a DVI line (atmospheric containment pressure)
- the double-ended rupture of a DVI line (138 kPa (20 psia) containment pressure)
- a cold-leg break of 25.4 cm (10 in.) equivalent diameter (atmospheric containment pressure)

NOTRUMP did not calculate any of these breaks 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 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 which that portion of the core that might experience critical heat flux was allowed to heat adiabatically until the combined flow from the two accumulators reduced the core void fraction. This calculation resulted in a PCT of 744 °C (1370 °F), which is much less than the 1204 °C (2200 °F) limit in 10 CFR 50.46. For break sizes larger than 25.4 cm (10 in.) equivalent diameter, even more core voiding and core heatup would be expected. The large break sizes, evaluated in Section 15.2.6.5.2 of this report, would bound the breaks discussed above.

In the AP1000, the hot-legs enter the RV at a lower elevation than do the cold-legs. A small break in a hot-leg might lead to a lower RV inventory than a break in a cold-leg of the same size. In RAI 440.098, the staff requested the applicant to perform additional SBLOCA analyses, including hot-leg breaks, for the AP1000 design. In response, the applicant provided the NOTRUMP predictions for the following small breaks:

- a 5.08-cm (2-in.) cold-leg break in the loop with the pressurizer (atmospheric containment pressure)

- a 5.08-cm (2-in.) hot-leg break in the loop without the pressurizer (atmospheric containment pressure)

- the double-ended break of a cold-leg pressure balance line to a CMT (atmospheric containment pressure)

None of these break sizes resulted in core uncovery. The staff concludes that the applicant has evaluated a sufficient small-break spectrum.

The double-ended severance of a DVI line represents a limiting sequence for SBLOCA analysis, because the water from one of the two accumulators, one of the two CMTs, and one of the two IRWST injection lines would not reach the RV, but would spill into the containment. For this reason, the staff concentrated much of the review effort on this postulated accident.

The staff has reservations on the ability of the NOTRUMP code to conservatively predict liquid entrainment within the upper plenum, hot-legs, and ADS-4 valves. If too little liquid entrainment were assumed, this liquid would be available to flow back into the core and provide unrealistic core cooling, and depressurization of the reactor system by the ADS-4 would be artificially enhanced. Both these effects would not be conservative for safety analysis. Westinghouse addressed this concern by performing an analysis for the double-ended DVI line break in which all liquid leaving the core was set at the same velocity as that of the steam (homogeneous flow). Using the homogeneous flow assumption, all liquid which reached the upper plenum would be swept out toward the ADS-4. The homogeneous analysis did not predict significant core uncovery. However, it predicted a lower minimum core water mass, compared to the nonhomogeneous case. The homogeneous analysis for a postulated double-ended DVI line break is part of the small break evaluation model for the AP1000 design.

As part of the validation of NOTRUMP, as discussed in Section 21.6.2 of this report, Westinghouse compared NOTRUMP predications with test data from the Advanced Plant Experiment (APEX)-1000 test facility. For most phenomena, the code compared well with the test data. However, the code was found to be nonconservative for prediction of water in the core in the early part of the tests simulating double-ended DVI line breaks. For this reason, the applicant performed an analysis of the double-ended DVI line break for the AP1000 using the Chang critical heat flux correlation (Chang, S.H., et al., “A Study of Critical Heat Flux for Low Flow of Water in Vertical Round Tubes Under Low Pressure,” <u>Nuclear Engineering and Design</u>, July 1991). The Chang correlation analysis demonstrated that the core will remain cooled during this period. Use of the Chang correlation to demonstrate core cooling for the double-ended DVI line break is part of the small break evaluation model for the AP1000 design.

In performing the double-ended DVI break analyses (including the homogeneous assumption analysis), the applicant took credit for the ADS-4 and the elevated containment pressure that would exist as a result of the energy added to the containment atmosphere by the break. Use of an elevated pressure increases the relieving capacity of the ADS-4 and shortens the time before IRWST injection begins. The applicant believes that 137.9 kPa (20 psia) is the minimum pressure that will occur within the containment building following the occurrence of a double-ended DVI line break during the time period covered by the NOTRUMP analysis. In calculating the minimum containment pressure, the applicant used the WGOTHIC code, which the NRC staff has reviewed, as discussed in Section 21.6.5 of this report. After discussions with the NRC staff, the applicant made a series of conservative assumptions for computing low containment pressures. The AP1000 minimum pressure and LTC models incorporate the following conservative assumptions, with the detailed maximum pressure model as the reference point:

- The containment net volume was increased by a factor of 1.1.
- The containment shell and the passive containment cooling system (PCS) heat structure areas were increased by a factor of 1.1.
- The remaining heat structure areas were increased by a factor of 2.1.
- The Uchida correlation with a multiplier of 1.2 was used for passive heat structures (non-PCS structures) throughout the accident.
- The PCS heat and mass transfer correlation multipliers were appropriately biased to account for the uncertainty in the experiential database, and forced convection was not included on the PCS inner surface.
- Heat transfer in dead-ended compartments below the operating deck was not turned off at the end of blowdown.
- The air gap between the steel and the concrete was reduced to zero, from the 20-mil thickness used in the maximum pressure calculation.
- The material properties for steel, concrete, air, and the inorganic zinc coating were biased high for conservatism.
- Heat transfer credit for the PCS was set to start at the beginning of the accident, earlier than was assumed for the maximum pressure calculation.
- Westinghouse maintained its treatment of ECCS spillage as implemented in 1979 (with the acceptance of the WGOTHIC breakflow model for this evaluation).
- The containment purge system was assumed to be operating at the start of the accident and isolated on a high-pressure signal.
- The initial and boundary conditions for the containment, the PCS water, and the environment were set to minimize the calculated pressure.

- The operators were assumed to actuate the non-safety-related air coolers 10 minutes after the break occurred.

Westinghouse used similar assumptions to compute the minimum containment pressures for ECCS evaluation of operating plants. The NRC staff reviewed the minimum back pressure calculation and concludes that they are an acceptable adaptation of the guidelines of SRP Section 6.2.1.5, “Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Studies,” and that the results are acceptable for use in the ECCS evaluation for the AP1000 design.

Although the core was predicted to remain covered following a double-ended DVI line break, even if the containment pressure is maintained at 101.4 kPa (14.7 psia) as a boundary condition for the NOTRUMP calculation, the applicant considered the 137.9 kPa (20 psia) containment back pressure boundary condition case to be the base case for the design basis. The LTC analysis for the design-basis double-ended DVI line break using WCOBRA/TRAC is initialized from the NOTRUMP analysis using the 137.9 kPa (20 psia) back pressure.

As an additional check on the NOTRUMP results obtained by the applicant, the staff performed a series of audit calculations of SBLOCAs for the AP1000 using the RELAP5 computer code. The staff developed RELAP5, an advanced T-H simulation tool. The RELAP5 analyses used conservative assumptions similar to those used by the applicant in the NOTRUMP analyses. Decay heat was set at 120 percent of the ANS-5.1-1973 Standard. The ANS-5.1-1973 decay heat standard is equivalent to the ANS-5.1-1971 standard used in the NOTRUMP analyses. 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 assessment of the possibility of fuel cladding heatup following a DNB condition or core uncovery event.

The staff performed audit calculations for the following cases:

- the inadvertent opening of both 10.16-cm (4-in.) ADS-1 valves

- a cold-leg break of 5.08-cm (2-in.) equivalent diameter in the loop without the pressurizer

- a cold-leg break of 8.89-cm (3.5-in.) equivalent diameter in the loop without the pressurizer

- 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 DVI line

- a cold-leg break of 25.4-cm (10-in.) equivalent diameter

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

Operating PWRs do not have ADS valves or a PRHR to depressurize and cool the RCS 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 SG tubes might flow into the lower cold-leg elevations and into the core. The water derived from steam condensation would not be borated, so that the entry of this water into the core might cause an increase in core power. The NRC staff does not believe deboration will occur during an SBLOCA for the AP1000 design. Section 15.2.8 of this report evaluates this issue.

Based on the foregoing considerations, the staff concludes that the applicant's analyses for a spectrum of small piping breaks in the reactor pressure boundary are acceptable and meet the requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50, and that the calculated performance of the passive ECCS 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-12945. WCOBRA/TRAC is the Westinghouse's BE T-H computer code used to calculate T-H conditions in the reactor system during blowdown and reflood of a postulated LBLOCA. This code consists of the BE features needed to satisfy the requirements of 10 CFR 50.46(a)(1)(i) for a realistic code.

In addition, the applicant used WCOBRA/TRAC to analyze the post-LOCA LTC of the AP1000, including decay heat assumptions from 10 CFR Part 50, Appendix K. Section 15.2.7 of this report discusses the staff's evaluation and conclusions for the LTC.

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 the 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, to calculate the 95th percentile PCT. The results of the analysis show that the double-ended cold-leg guillotine 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 the search for the limiting LBLOCA included the hot-leg break and the cold-leg limiting split break.

The applicant considered the 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 presented the results of the LBLOCA analyses in DCD Tier 2, Tables 15.6.5-5 through 15.6.5-8, and Figures 15.6.5.4A-1 through 15.6.5.4A-12. The applicant submitted additional information in its response to RAI 440.097, Revision 1, Table 15.6.5-8, and Revision 1 response to RAI 440.097, Figures 440.097R1-1 through 440.097R1-3. Following an LBLOCA, the reactor trip actuates on a low-pressurizer pressure trip signal. The LBLOCA analysis does not credit the insertion of the control rods. Within a few seconds after the initiation of an 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 an 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 used the WCOBRA/TRAC models to perform the LBLOCA analyses with calculated PCT uncertainties derived from the effects of model-related parameters, while the initial condition-related parameters used in the analyses are bounding and conservative.

The applicant addressed the limitations in WCOBRA/TRAC relating to the PCT for values greater than 940 °C (1725 °F). Staff review of the sensitivity calculations required by the code limitation indicates 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 LBLOCA analysis shows a high level of probability that the following criteria will be met:

- The calculated PCT will not exceed 1204 °C (2200 °F).

- The calculated maximum cladding oxidation will not exceed 0.17 percent of 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 1 percent of the amount that would be generated if the entire cladding metal surrounding the fuel (excluding the cladding surrounding the plenum volume) were 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.

The calculated results of the AP1000 LBLOCA satisfy the requirements of 10 CFR 50.46 and, therefore, are acceptable.

### 15.2.7 Post-LOCA Long-Term Cooling (DCD Tier 2, Section 15.6.5.4C)

This analysis establishes that (1) the core remains cooled for the duration of the LTC phase, (2) the boron concentration in the core keeps the core noncritical, and (3) boron precipitation will not obstruct core coolant flow.

15.2.7.1 Double-Ended Direct Vessel Injection Line Break

The applicant selected a double-ended guillotine break of the DVI line for the LTC evaluation. The double-ended DVI (DEDVI) line break is the most limiting LTC case, in the sense that it maximizes the decay heat generation rate and minimizes the cooling water injection head. The case analyzed is the continuation of an SBLOCA. Maximum design resistance is applied for the ADS-4 flowpath, the IRWST injection, and the containment recirculation. Failure of one of the two ADS-4 valves in the PRHR loop is assumed. ADS-1, 2, and 3 valves are not modeled here, because they do not practically impact depressurization with the ADS-4 valves opened.

The applicant used WCOBRA/TRAC for the LTC analysis. As described in Section 21.6.4 of this report, the staff has evaluated WCOBRA/TRAC and found it to be acceptable for the AP1000 LTC analysis. The LTC analysis uses a detailed nodalization model, which includes 4 radial channels and 17 axial nodes in each channel to represent the AP1000 core.

Initial conditions of the RCS liquid inventory and temperatures are taken from the NOTRUMP DEDVI case to initiate the calculation shortly after IRWST injection begins, and it proceeds until achievement of a quasi-steady state. At this time, the calculated results are independent of the initial conditions. The WCOBRA/TRAC calculation proceeds using boundary conditions from a corresponding WGOTHIC analysis and is carried to 10,000 seconds, until establishment of a quasi-steady-state sump recirculation. The minimum containment flood level for this LTC transient is 32.86 m (107.8 ft)(from the plant reference elevation), which is sufficient to inject coolant into the RV through the broken DVI line. Likewise, the IRWST also provides sufficient head to inject water through the intact DVI nozzle.

In the downcomer, the water level is about 5.49 m (18 ft)(from the bottom of the core), while the core collapsed level is about 2.44 m (8 ft). Boiling in the core produces steam and a two-phase mixture, which flows into the upper plenum. The boiling process, coupled with the changes in the steam quality exiting the ADS-4 lines, causes pressure variations, which in turn cause liquid flow variations entering the bottom of the core. This also results in liquid and vapor flow variations at the top of the core. The void fraction at the uppermost two nodes is about 0.8. Two-phase mixture continuously flows into the hot-legs, which, on average, have a collapsed liquid level of more than 50 percent. The exit flow rates through the pressurizer- and nonpressurizer-side ADS-4 valves are about 22.68 kg/s and 45.36 kg/s (50 lbm/s and 100 lbm/s), respectively. The broken DVI line indicates a small outward flow at the beginning of the transient, which reverses and reaches about 22.68 kg/s (50 lbm/s) at the quasi-steady state. The intact DVI line injects about 77.11 kg/s (170 lbm/s) at the start of the transient and diminishes to about 29.48 kg/s (65 lbm/s) at the quasi-steady state. The WCOBRA/TRAC analysis demonstrated that the void fraction in the top node is 0.8, and therefore the core is wetted. The peak clad temperature for the hot rod follows saturation temperature within -12.2 °C (10 °F), indicating that core uncovery and temperature excursion do not occur during LTC.

A variant of this transient for a containment pressure of 101.4 kPa (14.7 psia) was calculated using the window method and starting at 6500 seconds. The lower containment pressure creates a higher volume of steam for the same decay heat. The reduction in containment pressure coincides with the switchover to containment recirculation. The results also show that the core remains covered, with PCTs remaining very near the saturation level.

15.2.7.2 DEDVI Break and Wall-to-Wall Floodup—Containment Recirculation

The case of DEDVI break and wall-to-wall floodup with containment recirculation is a more limiting case, in that it assumes that all of the spaces beneath the sump level are flooded, corresponding to the minimum possible water level and the minimum injection head through the broken DVI line. Flooding of the dry compartments is conservatively estimated to take 14 days. The decay heat source is estimated for this time. DCD Tier 2, Section 15.6.5.4C.2, describes the calculations for the initial conditions for the window at floodup conditions, including the assumptions for an ADS-4 failure and 101.4 kPa (14.7 psia) containment pressure.

This LTC calculation was performed using the window mode methodology. The calculation was performed with the continuous mode until establishment of a quasi-steady state at about 400 seconds of transient time. The downcomer collapsed liquid level varies from 7.0 m to 7.6 m (23 ft to 25 ft), while the collapsed liquid level in the vessel is about 4 m (13 ft). Pressure spikes from the boiling process and changes in steam quality out of the ADS-4 lines cause injection liquid flow rate variations through the broken and intact DVI lines of an average value of approximately 22.68 kg/s (50 lbm/s). The top two nodes have average void fractions of about 0.8 and 0.7, respectively. The collapsed liquid level in the hot-leg is about .5 m (1.7 ft), while the peak clad temperature of the hot rod remains close to the saturation temperature. The vapor mass flow rate out of the core is about 1.6 kg/s (3.5 lbm/s), and the liquid flow rate is about 45.36 kg/s (100 lbm/s). This liquid flow out of the vessel is more than sufficient to remove excess boron. As described in Section 15.2.7.6 of this report, the staff also performed an independent evaluation of the AP1000 LTC behavior, which concludes that the boric acid concentration will not reach precipitation limits. In summary, the wall-to-wall flooding case with the lowest injection head keeps the core cooled and provides more than adequate liquid flow to preclude boron concentration in the core.

15.2.7.3 Post Accident Boron Concentration

Westinghouse performed an evaluation of the potential for boron concentration to build up in the core following a LOCA and during LTC. The evaluation considered short- and long-term intervals which correspond to the times before and after ADS actuation. In the short term, the time is relatively limited and continuous flow exists. For boron to concentrate in the vessel, significant amounts of borated water must be dumped into the vessel. This takes place when the cold-legs void, and the CMTs begin to inject. At a CMT level of 20 percent, the ADS-4 actuates. In this time interval, favorable conditions for boron concentration do not occur. The applicant's calculations show that a 2-in. break LOCA requires less than 16 minutes until the ADS is actuated. For larger breaks, the time is shorter. At the decay heat levels of the AP1000, the buildup of significant concentration requires more than 3 hours. In the long-term interval, once the ADS-4 is actuated, liquid outflow limits the boron concentration in the core. These calculations indicate that ADS-4 vent quality at the initiation of recirculation is about 50 percent and decreases to less than 10 percent at the time of wall-to-wall flooding. At the maximum vent quality, boron concentration is about 7400 ppm. The maximum boron solubility temperature is 14.4 °C (58 °F) (at 7400 ppm). The vessel is not expected to reach this temperature. Early in the LTC phase, with high decay heat, the high vent flow velocities can support an annular flow regime that moves water up and out the ADS-4 vent. In this case, a larger amount of water is expelled, creating a lower boron concentration in the core and a solubility temperature lower than 14.4 °C (58 °F). In summary, both physical considerations and calculation results indicate that the boron concentration in the core will not reach precipitation limits. The results of the AP1000 LTC phase are acceptable because boric acid precipitation in the core is precluded, preventing both criticality and/or flow blockage.

Operator Actions

The water level in the AP1000 sump varies from 33.3 m (109.3 ft)(from the plant reference elevation) at the beginning of recirculation for a non-DVI LOCA, to 31.5 m (103.5 ft) for a DVI LOCA and wall-to-wall flooding. Westinghouse stated that, during recirculation, the operators will be instructed to maintain the sump water level at or above the 32.6 m (107 ft) level. This practice adds a measure of assurance that adequate recirculation hydraulic head will exist during the LTC phase. Therefore, the core will remain covered with a two-phase mixture, and sufficient liquid entrainment will occur to ensure that boric acid is maintained well below precipitation limits.

Summary and Conclusion—LTC Boron Precipitation

Using WCOBRA/TRAC, Westinghouse analyzed the AP1000 LTC for the limiting case (i.e., highest decay heat level) DVI line break LOCA. Westinghouse also analyzed a variation of this case, which results in the lowest sump injection head through the broken DVI line (the wall-to-wall floodup case). The results show that sufficient ventflow (of medium- to low-quality steam) through the ADS-4 exists to remove enough water from the core to keep the boron concentration below 7400 ppm. The corresponding boron precipitation temperature is about 14.4 °C (58 °F), which is virtually unattainable in the vessel or the ADS-4 vent pipe. For the floodup case, the minimum sump injection head (through the broken DVI line) is adequate to maintain cooling and limit boron concentration. The applicant used an acceptable code to perform the analysis, and the results showed that the core boron concentration during LTC will not allow the core to become critical, nor will boron precipitation obstruct core coolant flow.

15.2.7.4 Additional Calculations

In Appendix H to WCAP-15644-P, Westinghouse provided additional analysis to (1) support the conclusion above regarding LTC boron precipitation and (2) to address a staff question regarding the instances with low decay heat that may not generate sufficient steam to support boron removal by expulsion of liquid through the ADS-4 valves. This section reviews this analysis.

The study consists of hand calculations and is based on a number of bounding assumptions. For the short term (as defined in Section 15.2.7.3 of this report), this study extends the size of the LOCA break to less than 5 cm (2 in.), with maximum makeup from the BAT. In such a case, the ADS may not actuate, and the plant would remain in this condition for some length of time. The hot-leg voids in about an hour, during which the CMT injects and increases the boron concentration. The BAT continues to inject borated water until it empties. Under those conditions, the maximum concentration is about 39,000 ppm, which could be reached in about 7 hours. In this case, the operator will likely be able to cool down and depressurize the RCS and avoid ADS actuation. The RCS temperature will remain well above the boron solubility temperature of about 87.8 °C (190 °F) (39,000 ppm) until the RNS cut-in temperature of 176.7 °C (350 °F) is reached. The RNS shutdown cooling operation would promote the RCS boron mixing. On the other hand, should the failure of the operator to shut the plant down actuate the ADS, injection from the IRWST will enter to dilute the boron concentration. In either case, the AP1000 emergency response guidelines (ERGs) require that the RCS be sampled for boron concentration to assure that sufficient shutdown margin exists and to prevent excessive concentration.

For the long term, after ADS-4 activation, boron concentration depends on the amount of water exiting from the ADS-4. A simplified model was developed, which considered the possibility of 100-percent quality steam through the ADS-4, or varying steam quality until the pressure drop through the system equals the driving injection head. In the first case, the hot water level will increase to the point where slugflow will be established. In the second case, the steam quality is sufficiently low so as to maintain an acceptable boron concentration. In addition, the method was used to estimate the capability of the core with reduced decay heat to expel water during LTC through the ADS-4. Such instances include (1) a very low core decay heat generation rate during the very long term and (2) the core decay heat immediately after refueling.

According to these results, with an initial core water level of 33.2 m (109 ft) and reasonable operator action to restore the sump level, the LTC phase can be maintained for a significantly long period of time (Westinghouse estimates this duration to be 6 years). Eventually, the decay heat cannot sustain steam velocity or quality to expel water. On the other end (i.e., a very short irradiation history, for which decay heat diminishes rapidly), a very small probability exists that a LOCA will occur during the time the reactor core has been operational. However, if a fresh core experiences a LOCA, the calculation indicates that the operator has adequate time to intervene and add water to restore the sump water level.

In summary, the auxiliary simplified calculations support the conclusions derived with WCOBRA/TRAC and extend the conclusion into low decay heat cases. The staff finds that emergency operating procedures and operator action can assure LTC for extended periods of time.

15.2.7.5 Closure of Open Item 15.2.7-1

In Revision 2 to the AP1000 DCD, the WCOBRA/TRAC LTC analysis was performed with the same noding as in the AP600 LTC analysis, which had only two axial nodes to represent the reactor core. However, the increased decay heat level and the increased core height to 4.3 m (14 ft) in the AP1000 design prompted the staff to request an analysis with a nodalization of sufficient detail to capture the spatial variation in the core void distribution. In the DSER, the staff identified Open Item 15.2.7-1, which stated that the applicant should provide more nodes in the vessel axial void distribution analysis during LTC to demonstrate that no possibility of core uncovery and adiabatic fuel heatup exists.

In the later revisions to the DCD, the applicant performed the LTC analysis, described in DCD Tier 2, Section 15.6.5.4C, based on a detailed nodalization model with 4 radial channels and 17 axial nodes to represent the reactor core. The LTC analysis results with this more detailed noding scheme were the subject of the preceding review and show that no long-term core uncovery occurred. The staff concludes that Open Item 15.1.7-1 is closed.

15.2.7.6 Staff Independent Post-LOCA Long-Term Cooling Calculation

This section describes the staff's independent calculation of the post-LOCA LTC behavior in the RCS to address the AP1000 system performance to maintain core cooling and preclude boron precipitation. During the LTC phase, the RCS will boil for extended periods of time, causing the boric acid concentration to increase in the vessel core and upper plenum regions.

Unlike conventional nuclear steam supply systems, the AP1000 does not rely primarily on operator action to control boric acid and prevent precipitation. Because of ADS-4 actuation, the AP1000 design ensures that boric acid is flushed from the core and RCS during the long term, because of the sustained entrainment of liquid from the upper plenum and hot-legs that is expelled through the ADS-4 piping to the containment. In order to evaluate LTC system performance, the staff developed several methodologies to show the inherent margin in the AP1000 that ensures a continued core cooling and flushing of the boric acid from the vessel during the post-LOCA LTC phase. The following discusses LTC behavior, models, and analytical results dealing with the prevention of boric acid precipitation in the AP1000.

LTC and the prevention of boric acid precipitation is addressed for all break sizes, including the double-ended DVI line break, which results in the earliest draintime for the IRWST and, hence, the earliest start of recirculation from the sump (i.e., 2.4 hours). Because this break results in the earliest start of recirculation, decay heat will be the highest, maximizing the pressure drop across the ADS-4 lines. Furthermore, because liquid must always be expelled out the ADS-4 lines to control boric acid, it is also necessary to show that the two-phase level remains above the centerline of the hot-leg. With the two-phase level above the centerline of the hot-leg, the steam will entrain liquid from the hot-leg piping and carry it out the ADS-4 lines as long as steaming rates are 6.8 kg/s (15 lb/s) or higher. Moreover, as the evaluation below demonstrates, once the steaming rate drops below 6.8 kg/s (15 lb/s), the pressure drop across the ADS-4 decreases so that the two-phase level rises to very near the top of the hot-leg. In this condition, a two-phase mixture is expected to spill from the RCS to the containment from the ADS-4 lines, continuing the removal of boric acid from the RCS. The steaming rate of

6.8 kg/s (15 lb/s) occurs after about 14 days following a reactor trip from full-power operating conditions, assuming no subcooling of the entering sump injection.

To show that liquid is expelled from the ADS-4 lines, a model will be described that demonstrates that the fluid levels in the vessel remain in the top half of the hot-leg. This condition is necessary during the long term to assure that boric acid is flushed from the system. That is, it will also be shown that sufficient liquid is expelled to demonstrate that more liquid concentrate is removed than is accumulated in the core and upper plenum during extended boiling periods lasting for upwards of 14 days.

15.2.7.6.1 Long-Term Core Cooling Evaluation

To evaluate the AP1000 long-term core cooling behavior, the staff developed a T-H model, which consists of computing (1) the pressure drop from the upper plenum through the hot-legs and the ADS-4 lines to the containment when single phase steam exits the RCS (maximized pressure drop for a given mass flow rate), and (2) the hydrostatic differential head of water between the sump/downcomer and the inner vessel containing the core and upper plenum. The computation of this hydrostatic head differential assumes that the two-phase level in the vessel always remains at the centerline elevation of the hot-legs. This head differential is then compared to the pressure drop across the ADS-4 lines for a range of decay heat levels, down to and including conditions 14 days after initiation of the LOCA. Thus, if the pressure drop across the ADS-4 lines is less than or equal to the hydrostatic head differential between the sump/downcomer and inner vessel, then there is assurance that the two-phase level will always remain in the top half of the hot-leg, thereby assuring entrainment of liquid and the removal of boric acid from the system.

The staff evaluation made the following bounding assumptions:

- The containment pressure is 101.4 kPa (14.7 psia).
- The water in the sump and downcomer is saturated at 101.4 kPa (14.7 psia).
- The sump water level is at Elevation 31.4 m (103 ft) at the start of recirculation, which is the minimum level or elevation calculated by Westinghouse to be at Elevation 31.5 m (103'-5") at 14 days.
- Recirculation begins at 8640 seconds (0.104 days).
- Decay heat generation is based on the ANS-5.1-1971 standard increased by 20 percent

The staff used a drift-flux model to compute the void distribution in the core and upper plenum regions of the inner vessel. Appendix A to technical report, ISL-NSAD-NRC-01-003, “Preliminary Results of the AP1000 RELAP5/MOD3.3 Analysis for the Two-Inch Cold Leg and Main Steam Line Breaks,” V. Palazov and L. Ward, Information Systems Laboratories, Inc., August 2001, describes this model in detail. This model was benchmarked against the thermal-hydraulic test facility (THTF) bundle uncovery tests and was validated against the ACHILLES low-pressure two-phase-level swell tests, as well as the THETIS boiloff tests, also performed at low pressure. The benchmarks showed that the drift-flux model predicted the void distribution (swelled level) within the experimental uncertainty for the tests.

To compute the pressure drop across the ADS-4 lines from the upper plenum at the two-phase surface, the staff developed an additional methodology. This model consisted of a simultaneous solution of the mass, energy, and momentum equations employing a volume-flowpath-network arrangement using a semi-implicit numerical scheme. The region from the upper plenum through each of the hot-legs and the ADS-4 lines was modeled as a series of parallel volumes and connecting flowpaths or junctions. The solution to the conservation equations produced a simultaneous solution of both the system pressure distribution and attendant mass flow rates. The equation for the time rate of change of pressure at each volume was derived from the definition of specific volume of the fluid, the energy equation, and the equation of state. The momentum equation contained the inertia term, the static pressure drop between cells, the geometric and friction pressure drop, and the recoverable momentum flux acceleration pressure losses. The time rate of change equations for pressure and momentum form a coupled set of differential equations. Westinghouse provided the geometric K-factors used to represent the piping and components from the upper plenum through the ADS-4 lines to the containment. The staff model increased the frictional pressure losses by 10 percent. In addition, the least-resistant ADS-4 line was conservatively assumed to fail closed.

The model was applied to the pressure drop data and sample problems in Crane ("Flow of Fluids Through Valves, Fittings, and Pipes," Crane Co., Technical Paper No. 14, 1988) and shown to reproduce the pressure drop for steam flowing in pipes with various geometric and frictional pressure losses. The model was also compared to the data of J.K. Ferrel, et al., (Ferrel, J.K., "Two-Phase Flow through Abrupt Expansions and Contractions," TID-23394 (Volume 3), North Carolina State University, June 1966) and predicted the pressure loss across contractions for a range of area ratios. Furthermore, this method was compared to a range of pressures and ADS-4 steam flow rates, and it was also shown to reproduce the pressure loss across the ADS-4 lines computed by Westinghouse using its detailed FLOAD4 network code.

Using the staff methodology described above, Figure 15.2.7.6 of this report compares the calculated hydrostatic head differentials between the sump/downcomer and inner vessel, covering the range of decay heat steaming rates from 25 kg/s to 5 kg/s (54.1 lb/s to 10.6 lb/s) (1.4 h to 14 days). The start of recirculation is 8640 seconds (2.4 h), with a steaming rate in the core of 21 kg/s (46.1 lb/s). Figure 15.2.7.6 of this report compares the head differential to the pressure drop across the ADS-4 lines over these decay heat generation rates. The two-phase level would be at the hot-leg centerline elevation only when the pressure loss across the ADS-4 lines equals the differential head of water between the sump/downcomer and inner vessel region. Clearly, when the ADS-4 pressure loss exceeds the differential head curve in Figure 15.2.7.6 of this report, the pressure loss across the ADS-4 lines is high enough to depress the two-phase level below the centerline of the hot-leg. In Figure 15.2.7.6 of this report, the vertical line indicates this condition so that, for steaming rates of about 23 kg/s (50 lb/s) and higher, the two-phase level will be depressed below the hot-leg centerline if only pure steam (quality = 1.0) should exit the system. The region to the right of the vertical line demonstrates that when only single vapor exits the ADS-4 lines, the pressure loss across the ADS-4 lines is insufficient to depress the two-phase level below the centerline of the hot-leg.

This demonstrates that the two-phase level would be higher than the prescribed two-phase level, which is always assumed to be at the hot-leg centerline. Because entrainment will remove liquid from the hot-legs for steaming rates above 7 kg/s (15 lb/s) (approximately 14 days after reactor trip), liquid is expelled from the RCS for 14 days following a LOCA. In fact, the difference between the upper and lower curve represents the excess margin in the AP1000 design and can be interpreted to mean that the two-phase level will remain near the top of the hot-leg from the earliest time for start of recirculation at 2.4 hours, through to decay heat levels producing steaming rates down to 7 kg/s (15 lb/s), or 14 days after reactor trip.

Furthermore, if the minimum sump level computed by Westinghouse based on conservative assumptions regarding containment performance is used, even more margin is available to guarantee the expulsion of liquid from the RCS. Bounding containment calculations by Westinghouse show that the sump level at 2.4 hours is at Elevation 32.9 m (108 ft) in the containment (higher than the assumed Elevation 31.4 m (103 ft) in the staff analysis). Using the sump level of 32.9 m (108 ft) at 2.4 hours which decreases to 32.6 m (107 ft) at 1 day, as shown in Figure 15.2.7.6 of this report, at the earliest time of recirculation (2.4 h), a margin of approximately 20.7-kPa (3-psi) in the pressure loss is required to depress the two-phase level below the hot-leg centerline. This water is also subcooled, but the Figure 15.2.7.6 analyses did not credit it. Subcooling would further increase the excess margin, as would higher containment pressures, because at 2.4 hours the minimum containment pressure is 131 kPa (19 psia). At these low pressures, an increase from the assumed 101.4 kPa (14.7 psia) containment pressure to the minimum justified pressure of 131 kPa (19 psia) would further reduce the pressure loss across the ADS-4 lines, since the pressure drop would decrease because of the lower steam specific volume at the higher pressures. Clearly, excess margin exists to cover uncertainty in the pressure losses in the ADS-4 lines (uncertainty in the CRANE resistances and unknown variations in the line resistance set up possibly by the interactions between two closely connected elbows or bends), as well as any additional uncertainty that may exist in the line resistances connecting the sump to the downcomer. Figure 15.2.7.6 of this report shows that, at the start of recirculation (2.4 h), a 20.7 kPa (3 psi) margin exists. This estimate is conservative because it did not credit subcooling and neglected the benefit of the higher containment pressure of 131 kPa (19 psia).

Lastly, this excess margin can also accommodate compressible effects. Crane shows that a compressible factor of 1/0.87 should be applied to the pressure drop when the hot-leg is at 137.9 kPa (20 psia), and the steam flow rate through one hot-leg is 13.61 kg/s (30 lb/s). The excessive margin shown in Figure 15.2.7.6 of this report more than sufficiently covers increases in the pressure drop from compressibility effects.

In view of these results, sufficient margin in the AP1000 design exists to assure that liquid will be expelled from the RCS through the ADS-4 lines for as long as 14 days following reactor trip. In fact, because of the excess margin displayed in Figure 15.2.7.6 of this report, two-phase fluid would be expected to spill from the RCS through the ADS-4 lines well beyond the 14-day period illustrated in this graph. Even as late as 14 days, ERGs will require the operators to take actions to secure the plant. These actions include raising the sump level to the elevation between 33.2 m (109 ft) and 33.5 m (110 ft) by filling the containment sump with subcooled water from the RNS. This action would ensure liquid expulsion from the RCS for an extended period of time and would ensure that the steaming rate in the core would be terminated because of the subcooling of the sump water and the very low decay heat generation beyond 14 days. Limiting the sump level to lower than 33.5 m (110 ft) will preclude submergence of the ADS-4 valves, thus preventing core uncovery.

15.2.7.6.2 Boric Acid Precipitation Evaluation

The AP1000 does not require operator action to mechanically flush the boric acid from the RCS. For larger and intermediate breaks, which result in the actuation of ADS-4, the RCS entrains and expels sufficient liquid to maintain the boric acid content well below precipitation limits. For very small breaks where the RCS pressure remains high for extended periods of time to prevent ADS-4 actuation, ERGs will require the operators to depressurize the RCS or refill the RCS with charging injection to disperse the boric acid throughout the primary system. The staff evaluates both situations with regard to the boron precipitation concerns, beginning with the large and intermediate break system response, followed by an assessment of the very small breaks.

Larger and Intermediate Breaks

To assess boric acid buildup, the staff developed a simple model to compute the boric acid accumulation in the vessel over time. The model describes the boric acid content in the core and upper plenum as a function of time with and without the flushing of boron via ADS-4 liquid entrainment.

Since the DEDVI line break produces the earliest IRWST draintime and, hence, the earliest start of sump recirculation, this break is the limiting event and is used to investigate the buildup of boric acid. The first staff calculation credits no core flushing or entrainment out the ADS-4 lines, so that a bounding calculation is performed to identify the earliest time to precipitation. In essence, the no-flushing calculation is independent of break size because boric acid buildup depends on decay heat. The result of the no-flushing assumption shows that boron precipitation could occur about 5 hours into the event.

The second calculation assumes flushing through ADS-4 liquid entrainment. This analysis includes the following bounding assumptions:

- The decay heat generation is increased by 20 percent.
- The mixing volume is based only on the core region.
- The BAT injects at the maximum rate of 662.45 L/min (175 gpm).
- The concentration of the BAT is 2.4 percent.
- No entrainment or flushing is credited before IRWST drainage.
- At the start of recirculation, only a 2.27 kg/s (5 lb/s) flushing flow is assumed (flow in excess of the core boiloff rate).

The calculation credits ADS-4 liquid entrainment at the initiation of sump recirculation, which occurs 2.4 hours into the event. At 2.4 hours or the start of recirculation, a flushing flow out the ADS-4 lines of only 2.27 kg/s (5 lb/s) is assumed, which is the flow in excess of the decay heat steaming rate of about 20.87 kg/s (46 lb/s). The 2.27-kg/s (5-lb/s) liquid flow out of ADS-4 valves is a very conservative bounding assumption, as the RELAP5 calculations of the DEDVI line break show that the liquid flow out of the ADS-4 during the ADS-4 blowdown period and LTC period well exceeds 23.13 kg/s (51 lb/s). The calculation used containment pressures, sump levels, and sump temperatures from the bounding LTC analysis provided by Westinghouse as boundary conditions for the determination of the liquid flow out the ADS-4 lines.

With this bounding assumption, the calculated boric acid concentration in the core over time shows that at the start of recirculation, the excess flushing flow of 2.27 kg/s (5 lb/s) is sufficient to reduce the boric acid concentration in the vessel. Furthermore, during the blowdown or before IRWST injection, ADS-4 actuation would produce a liquid flow out the valves in excess of the boiloff, which would act to reduce the boric acid concentration much earlier in this event. Even with no credit for entrainment during blowdown, for conservatism, the boric acid concentration does not reach the precipitation limit of 32 wt% (corresponding to the saturation temperature at the minimum containment pressure of 137.9 kPa (20 psia) at the start of recirculation). This calculation produces a higher boric acid concentration than that calculated by Westinghouse. Calculations taking credit for a minimal flushing flow of 2.27 kg/s (5 lb/s) when the ADS-4 valves first open during blowdown (i.e., about 2000 seconds) show that the maximum concentration is 15.8 wt% (27,538 ppm), achieved shortly after the start of recirculation. This result assumes saturated inlet conditions to maximize the boiloff rate and the rate at which boron builds up in the core. When subcooling is credited at 48.9 °C (120 °F), the concentration remains well below the 27,538 ppm concentration because of the decreased boiloff rate in the core.

Based on this bounding analysis, the staff concludes that the boric acid concentration will not reach precipitation limits following breaks in the RCS that produce ADS-4 actuation.

<u>Very Small Breaks</u>

In the event of a very small break with a diameter of 1.3 cm (0.5 in.) or less, the RCS could boil for extended periods of time while remaining at elevated pressures and conditions, which would delay actuation of the ADS-4 valves. Under these conditions, injection from the CMTs, accumulators, and BAT could increase the boric acid concentration in the vessel. Since the break is small with RCS pressures remaining high, the CMTs do not rapidly drain, delaying the ADS-4 valve actuation. If boiling persists for several hours, concentrations could increase to values that, although soluble at the high pressures and temperatures, could cause precipitation should ADS-4 eventually actuate and rapidly depressurize the RCS. To investigate the time to reach precipitation limits corresponding to RCS pressures at 137.9 kPa (20 psia), the staff performed an analysis to determine the time required to increase the boric acid content in the core to 32 wt%. Assuming the BAT injects with the boric acid content of 2.4 percent, about 3.5 hours of boiling would be sufficient to increase the boric acid content in the core to 32 wt%. This assumption is considered to be bounding because the staff assumed saturated inlet conditions and did not credit the fact that, as subcooled water enters the core, the boiloff rate and the rate at which boric acid builds up in the core are significantly reduced.

As documented by Westinghouse, ERGs will instruct the operators to sample RCS boric acid content and take actions to reduce the concentration of the injected boric acid or initiate an early cooldown to prevent boric acid precipitation, in anticipation of ADS-4 actuation after extended periods of boiling. The ERGs assure that precipitation limits are never exceeded following very small breaks in the RCS. Such actions to control boric acid for the very small breaks are directed after 1 hour, but no later than 3 hours.

### 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 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 because the AP1000 does not rely on SGs to cool the RCS during an SBLOCA event. Consequently, the SGs should not generate any significant amount of conservatively assumed boron-free condensate via reflux condensation over an extended period of time during an SBLOCA event. The following describes the staff evaluation in detail.

During an 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 HX becomes a dominant RCS heat sink following the generation of an "S" signal during a postulated SBLOCA event. As such, the PRHR HX could become a potential source for generating a volume of unborated coolant during an SBLOCA. Such a scenario could lead to a reactivity excursion as a result of a restart of an 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 design for the following reasons. Specifically, the AP1000 reactor coolant loop piping does not contain a loop seal, and thus no point exists to collect a large slug of unborated condensate in the reactor coolant loop piping. During the SBLOCA event, once subcooling in the RCS is lost, steam will enter the PRHR HX and will condense on the inside of the PRHR HX 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 RV.

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 CMTs, 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, which show that boron dilution from the operation of the PRHR HX 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 the BOC of approximately 1000 ppm. The low AP1000 core boron concentration significantly reduces the potential for the PRHR to dilute the coolant in the RV to the point of criticality. Although the AP1000 PRHR flow rate is high, the CMT flow rate, the RV downcomer, and the lower plenum volume for the AP1000 are also high. Taking these differences into account, the AP1000 design studies show that post-LOCA boron dilution is not a concern, provided that good mixing exists in the vessel. Analysis for the AP1000 showed that mixing in the RV downcomer and lower plenum will counteract boron dilution in the core from PRHR operation.

NUREG/IA-0004, "Thermal Mixing Tests in a Semiannular Downcomer with Interacting Flows from Cold Legs," issued October 1986, reported the study of the mixing of high-pressure safety injection water with primary coolant in a simulated PWR downcomer. Test #106 in NUREG/IA-0004 considers a geometry which represents 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 RV 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. The Test #113 results in NUREG/IA-0004 provide further support for AP1000 downcomer mixing.

Test #113 was run at a simulated HPI rate that 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 enter 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 AP1000 DCD, and on the analysis performed by Westinghouse on behalf of this event, including the thermal tests mentioned in NUREG/1A-0004, the staff finds that the analysis in support of the possible deboration from an SBLOCA event is acceptable.

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

An ATWS event is defined as an AOO (such as LONF, 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 must be diverse from the existing RTS and must be capable of testing at power. This equipment will 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, 10 CFR 50.62(c)(1) specifies the basic requirements of the ATWS Rule:

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

The AP1000 design relies on the PRHR in lieu of an auxiliary or emergency feedwater system as its safety-related method for removing decay heat. In its letter, "AP1000 Request for Exemption," DCP/NRC-1534, dated December 3, 2002, the applicant submitted a request for exemption from the part of the ATWS Rule, 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 provided the results of an ATWS analysis in AP1000 Probabilistic Risk Assessment, Appendix A, Section A4, for the staff to review. In its analysis, the applicant used a complete 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 performed 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 equilibrium cycle core (ECC) conditions and one for first cycle core (FCC) conditions. The ECC case used an MTC of -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), which is less than the limit of 22.06 MPa (3200 psia). For the FCC case, the LONF analysis shows that with an MTC of -10 pcm/°F, the calculated peak RCS reaches 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 cycle when the reactivity feedback is not sufficient to maintain pressure under 22.06 MPa (3200 psia) for a given reactor state. The staff requested, in a followup question to RAI 440.014, that 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 performed a probabilistic risk assessment (PRA) evaluation to identify the frequency of the anticipated transients in DCD Tier 2, Chapter 15, and specify the most risk-significant ATWS scenarios for the AP1000 design. Based on the results of the PRA evaluation, the applicant identified the most risk-significant ATWS scenarios which make up more than 95 percent of the AP1000 ATWS initiating event frequency. The applicant performed ATWS analyses on the most risk-significant ATWS cases that, based on the results of the DCD Tier 2, Chapter 15, non-LOCA analysis for the AP1000, may result in a significant pressure increase during the transients. The applicant analyzed the following cases in detail to identify the scenario that results in the least margin to the RCPB 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
- LONF event with turbine bypass system operable
- LONF event without spray system, without turbine bypass system operable
- LONF event with turbine bypass system operable, more realistic SG heat transfer model
- complete loss of forced coolant flow with main feedwater system (MFWS) operable with turbine bypass system operable

- complete loss of forced coolant flow with MFWS operable without turbine bypass system operable

- complete loss of forced coolant flow induced by the loss of ac power, MFWS not operable, no steam dump operable, turbine trip at the initiation of the transient

The results of the ATWS analysis for the above cases confirm that, for the AP1000 design, the limiting case is the LONF 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 for 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-level signal. Together with implementation of a new DAS logic, an additional change has been implemented in the PLS, such that the PLS isolates the steam dump system whenever the SG level drops below the low wide-range SG water-level setpoint. DCD Tier 2, Section 7.1.1, includes the description of the new DAS logic.

Section 7.7.1 of this report discusses the review and acceptance of the new logic. The applicant performed an ATWS analysis of the LONF event for the AP1000, with the new DAS ATWS protection logic assuming an 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 LONF event, the maximum calculated RCS pressure is 19.43 MPa (2818 psia), which is within the acceptance limit of 22.06 MPa (3200 psia). The limiting ATWS case demonstrates that the UET (i.e., the time during the fuel cycle when the reactivity feedback is not sufficient to maintain pressure under 22.06 MPa (3200 psia) for a given reactor state) does not exist for the AP1000 design. RAI 440.014, Revision 1, and AP1000 Probabilistic Risk Assessment, Appendix A, Section A4, include the information on the new ATWS analysis discussed in this section. Because (1) the ATWS analysis used the LOFTRAN code previously approved by the NRC, (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 falls 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 concludes that the AP1000 design meets the acceptance criteria for these transients and accidents.

As discussed in Section 15.2.9 of this report, the staff concludes that the applicant’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 emergency feedwater system is not necessary to achieve the underlying purpose of 10 CFR 50.62(c)(1). Therefore, the exemption request is acceptable.

## 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 ( $\chi/Q$ values) provided in DCD Tier 2, Table 15A-5. Because all other aspects of the design are fixed, these $\chi/Q$ 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 include the following:

- MSLB outside containment (DCD Tier 2, Section 15.1.5)
- RCP shaft seizure (locked rotor) (DCD Tier 2, Section 15.3.3)
- RCCA 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 (FHA) (DCD Tier 2, Section 15.7.4)

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

- 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
- using a set of hypothetical atmospheric dispersion factor ( $\chi/Q$) values

The $\chi/Q$ 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 $\chi/Q$ values for the AP1000 design using meteorological data that is expected to bound 70 to 80 percent of U.S. operating nuclear power plant sites for offsite dispersion. In DCD Tier 2, Tables 2-1 and 15A-5 list the AP1000 hypothetical $\chi/Q$ values, and DCD Tier 2, Table 15A-6, lists the AP1000 hypothetical $\chi/Q$ values for the control room.

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 LPZ for the duration of exposure to the release cloud. The staff used a criterion of 0.05 Sv (5 rem) TEDE for evaluating the radiological consequences from DBAs in the control room of the 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 in its review of the AP1000 DBA radiological consequence analyses. RG 1.183 provides guidance on radiological consequence analyses to those licensees of operating power reactors choosing to implement an alternative source term pursuant to 10 CFR 50.67, which has the same regulatory dose criteria as those specified in 10 CFR 50.34(a)(1)(ii)(D) (i.e., 0.25 Sv (25 rem) TEDE) and GDC 19 (i.e., 0.05 Sv (5 rem) TEDE). Although RG 1.183 applies 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 the AP1000 is an advanced PWR.

Technical Evaluation

The staff reviewed the radiological consequence analyses performed by Westinghouse using the hypothetical $\chi/Q$ values given in DCD Tier 2, Tables 15A-5 and 15A-6. The staff 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 $\chi/Q$ 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 following sections describe the staff’s findings.

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 scenarios define the most severe accidents from which the plant could be expected to return to a safe-shutdown condition. The revised source terms in NUREG-1465 must be applied conservatively in evaluating DBAs in conjunction with conservative assumptions in calculating doses, such as adverse meteorology. Application to severe accidents may use more realistic assumptions.

The staff considered the inclusion of the ex-vessel and the late in-vessel source terms to be unduly conservative for DBA purposes. Such releases would only result from core damage accidents with vessel failure and core-concrete interactions. For passive ALWRs, the estimated frequencies of such scenarios are low enough that they do not have to be considered credible for the purpose of meeting 10 CFR 50.34. In SECY-94-302, 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.

The NRC issued RG 1.183 in July 2000 to provide guidance to licensees of operating power reactors on acceptable applications of alternative source terms pursuant to 10 CFR 50.67. This RG provides guidance based on insights from NUREG-1465 and significant attributes of other alternative source terms that the NRC staff may find acceptable for operating light-water reactors (LWRs). It also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted alternative source term for operating power reactors. The applicant followed the relevant guidance in RG 1.183 for PWRs.

Post-Accident Containment Water Chemistry Management

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

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

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

In the AP1000, the pH of the containment water will be between 7 and 9.5, which is the range found acceptable in Branch Technical Position (BTP) Materials Engineering Branch (MTEB) 6-1 of the SRP. A predetermined amount of trisodium phosphate (TSP) stored in the stainless steel baskets situated on the containment floor will maintain the pH in this range. After a LOCA, when the containment flooding water reaches the level of the baskets, the TSP will dissolve, and the solution of TSP will exercise buffering action and maintain sump water pH in the required range.

Acidic and basic chemical species released to the containment from different sources in the plant determine the pH of the containment sump water. Boric acid produces the most significant effect on reducing containment water pH. Westinghouse identified the RCS, IRWST, CMTs, accumulators, CVCS BAT, and spent fuel pool cooling system (SFS) cask loading pit as sources of boric acid. Westinghouse did not 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 the 30 days following a core damage accident include hydrochloric acid produced by radiolytic decomposition of electric cable jackets and nitric acid produced by radiolytic formation from air dissolved in the sump water. In addition, there is cesium hydroxide present, released from the damaged core. Because it is a strong base, cesium hydroxide will contribute to the increase of the pH. The applicant has determined that the baskets in the containment sump must store 12,492 kg (27,540 lb) of TSP to maintain the pH in the range of 7 to 9.5. While 7,503 kg (16,540 lb) of TSP is needed to neutralize boric acid, the rest will neutralize other acids existing in the containment and provide a 35-percent safety margin, which includes 10 percent 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 postaccident management of water chemistry in the AP1000 plant design. The staff finds that the amount of TSP stored in the containment sump will ensure a basic environment that will minimize the 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 4.

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 ( $\chi/Q$) values at fixed EAB and LPZ distances. DCD Tier 2, Tables 15A-5 and 15A-6 list the hypothetical reference $\chi/Q$ values used in the radiological consequence analyses for the AP1000 design. Section 2.3.4 of this report provides the staff's discussion of the hypothetical atmospheric dispersion factors. The staff will perform an independent assessment of short-term (less than or equal to 30 days) atmospheric dispersion factors for potential accident consequence analyses on a site-specific basis for a COL application that references the AP1000 design. If site-specific atmospheric dispersion factors exceed the reference values used in this evaluation (e.g., poorer dispersion characteristics), a COL applicant may have to consider compensatory measures, such as increasing the size of the site or providing additional ESF systems to meet the relevant dose limits set forth in 10 CFR 50.34 and GDC 19.

The staff had not completed its review of the control room atmospheric dispersion factors at the time the DSER was issued. Because the radiological consequence analyses for control room habitability use the control room atmospheric dispersion factors as an input, the staff could not make a finding on the acceptability of the Westinghouse analyses. In addition, the staff could not make a finding on whether Westinghouse had shown that the AP1000 design would be expected to meet GDC 19 for the hypothetical meteorological conditions used in the DCD. With the conclusion of the staff's review of the control room atmospheric dispersion factors, discussed in Section 2.3.4 of this report, the staff has completed its review of the control room habitability radiological consequences analyses. Therefore, DSER Open Item 15.3-2 is closed.

### 15.3.1 Radiological Consequences of a Main Steam Line Break Outside Containment

Both the staff and Westinghouse have evaluated the radiological consequences of a postulated SLB accident occurring outside of the containment and upstream of the MSIVs. 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 the following parameters:

- 567.8 L/d (150 gal/d) of primary-to-secondary leakage through any one SG, as specified in the AP1000 TS

- 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 (SFP) cooling capability. Section 15.3.9 of this report discusses the staff's review of the radiological consequences of SFP boiling.

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 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 indicated, and the staff agreed, that no DNB is expected to occur. Therefore, the calculation did not assume fuel-cladding failure. 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 MSLB accident. Before the accident, the AP1000 reactor was assumed to operate at the AP1000 TS equilibrium limit of 37 kBq/gm (1.0 µCi/gm) for 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) for DEI-131.

Tables 15.3-2 and 15.3-1 of this report provide the major parameters and assumptions used by the staff for the MSLB accident and the results of the staff's radiological consequence analyses for the EAB, LPZ, and control room, respectively. 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 SFP cooling capability will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 50.34.

The staff also 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) for DEI-131 and coincident with the loss of SFP cooling capability will not exceed the dose criterion set forth in 10 CFR 50.34 (i.e., 0.25 Sv (25 rem) TEDE).

In DCD Tier 2, Section 6.4.4, Westinghouse reported the results of its radiological consequence analysis for personnel in the main control room during a design-basis MSLB, relying on the main control room emergency habitability system (VES) to limit the radioactivity to which the personnel may be exposed. Section 6.4 of this report describes the staff's review and assessment of the VES. To verify the Westinghouse assessment, the staff performed an independent radiological consequence calculation for the MSLB accident with VES operation under high-high radiation levels. The staff finds reasonable assurance that the VES, under high-high radiological conditions as described in DCD Tier 2, Section 6.4, can mitigate the dose in the main control room following a design-basis MSLB to meet the dose criterion specified in GDC 19. Tables 15.3-9 and 15.3-1 of this report provide the major assumptions used by the staff and the resulting radiological consequence analysis results for the control room operators, respectively.

The calculation of the dose to the control room operators does not rely on the nuclear island nonradioactive ventilation system (VBS), a non-safety-related system, to meet the requirements of GDC 19. Under some accident circumstances, the non-safety-related VBS would be available to pressurize the control room and the technical support center (TSC) with filtered air and to provide recirculation cleanup. Section 9.4 of this report describes the staff's review and assessment of the non-safety-related VBS. In DCD Tier 2, Section 6.4.4, Westinghouse also reported the results of its radiological consequence analysis for personnel in the main control room and the TSC during a design-basis MSLB with the VBS available. The staff finds reasonable the Westinghouse assertion that, if available, the VBS can mitigate the dose in the main control room and the TSC following a design-basis MSLB to be within 0.05 Sv (5 rem) TEDE. Table 15.3-9 of this report provides the major assumptions used by the staff and the resulting radiological consequence analysis results for the personnel in the control room and the TSC with VBS available.

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

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, causes the reactor primary coolant pump seizure accident. Westinghouse analyzed this hypothetical accident assuming 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 to the reactor coolant. The maximum allowable 1135.6 L/d (300 gal/d) of primary-to-secondary leakage through two SGs, as specified in the AP1000 TS, carries the activity released to the primary coolant into the secondary coolant. The steamline safety valves or the PORVs release the activity to the environment. The applicant submitted a radiological analysis for the reactor primary coolant pump seizure accident in DCD Tier 2, Section 15.3.3.

The applicant also considered a coincident loss of SFP cooling capability. Section 15.3.9 of this report discusses the staff's review of the radiological consequences of SFP boiling.

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 assessment, the staff performed independent radiological consequence calculations for the reactor primary coolant pump seizure accident using the RG 1.183 source terms. Tables 15.3-3 and 15.3-1 of this report provide the major parameters and assumptions used by the staff and the results of the staff's radiological consequence analyses, respectively. 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 SFP cooling capability will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 50.34.

In DCD Tier 2, Section 6.4.4, Westinghouse reported the results of its radiological consequence analysis for personnel in the main control room during a design-basis locked rotor accident, relying on the VES to limit the radioactivity to which the personnel may be exposed. Section 6.4

of this report describes the staff's review and assessment of the VES. To verify the Westinghouse assessment, the staff performed an independent radiological consequence calculation for the locked rotor accident with VES operation under high-high radiation levels. The staff finds reasonable assurance that the VES, under high-high radiological conditions as described in DCD Tier 2, Section 6.4, can mitigate the dose in the main control room following a design-basis locked rotor accident to meet the dose criterion specified in GDC 19. Tables 15.3-9 and 15.3-1 of this report provide the major assumptions used by the staff and the resulting radiological consequence analysis results for the control room operators, respectively.

The calculation of the dose to the control room operators does not rely on the nuclear island VBS, a non-safety-related system, to meet the requirements of GDC 19. Under some accident circumstances, the non-safety-related VBS would be available to pressurize the control room and the TSC with filtered air and to provide recirculation cleanup. Section 9.4 of this report describes the staff's review and assessment of the non-safety-related VBS. In DCD Tier 2, Section 6.4.4, Westinghouse also reported the results of its radiological consequence analysis for personnel in the main control room and the TSC during a design-basis locked rotor accident with the VBS available. The staff finds reasonable the Westinghouse assertion that, if available, the VBS can mitigate the dose in the main control room and the TSC following a design-basis locked rotor accident to be within 0.05 Sv (5 rem) TEDE. Table 15.3-9 of this report provides the major assumptions used by the staff and the resulting radiological consequence analysis results for the personnel in the control room and the TSC with VBS available.

### 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 an 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. This mechanical failure causes 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 SFP cooling capability. Section 15.3.9 of this report discusses the staff's review of the radiological consequences of SFP boiling.

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.

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 assumed to leak into 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 1135.6 L/d (300 gal/d), as specified in the AP1000 TS. For both pathways, the applicant assumed that the AP1000 reactor operated 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 the RG 1.183 source terms. Tables 15.3-4 and 15.3-1 of this report provide the major parameters and assumptions used by the staff and the results of the staff's radiological consequence analyses, respectively. 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 RCCA ejection accident coincident with the loss of SFP cooling capability will fall well within the dose criterion set forth in 10 CFR 50.34 (i.e., 25 percent or 0.063 Sv (6.3 rem) TEDE).

In DCD Tier 2, Section 6.4.4, Westinghouse reported the results of its radiological consequence analysis for personnel in the main control room during a design-basis RCCA ejection accident, relying on the VES to limit the radioactivity to which the personnel may be exposed. Section 6.4 of this report describes the staff's review and assessment of the VES. To verify the Westinghouse assessment, the staff performed an independent radiological consequence calculation for the RCCA ejection accident with VES operation under high-high radiation levels. The staff finds reasonable assurance that the VES, under high-high radiological conditions as described in DCD Tier 2, Section 6.4, can mitigate the dose in the main control room following a design-basis RCCA ejection accident to meet the dose criterion specified in GDC 19. Tables 15.3-9 and 15.3-1 of this report provide the major assumptions used by the staff and the resulting radiological consequence analysis results for the control room operators, respectively.

The calculation of the dose to the control room operators does not rely on the nuclear island VBS, a non-safety-related system, to meet the requirements of GDC 19. Under some accident circumstances, the non-safety-related VBS would be available to pressurize the control room and the TSC with filtered air and to provide recirculation cleanup. Section 9.4 of this report describes the staff's review and assessment of the non-safety-related VBS. In DCD Tier 2, Section 6.4.4, Westinghouse also reported the results of its radiological consequence analysis for personnel in the main control room and the TSC during a design-basis RCCA ejection accident with the VBS available. The staff finds reasonable the Westinghouse assertion that, if available, the VBS can mitigate the dose in the main control room and the TSC following a design-basis RCCA ejection accident to be within 0.05 Sv (5 rem) TEDE. Table 15.3-9 of this report provides the major assumptions used by the staff and the resulting radiological consequence analysis results for the personnel in the control room and the TSC with VBS available.

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

GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment," 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 to perform necessary functions may be acceptable based on some other defined bases. For these lines, methods of mitigating the consequences of a rupture are necessary because the lines cannot be isolated automatically. For the AP1000 design, the two small lines in this category are

- the RCS sample line
- the discharge line from the CVCS to the liquid radwaste system (WLS)

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

When boron dilution operations generate excess primary coolant inventory, the CVCS purification flow is diverted out of containment to the WLS. Before passing outside containment, the flow stream passes through the CVCS HXs and mixed bed demineralizer for processing. The flow leaving the containment will be at a temperature of less than 60 °C (140 °F). The flow from a postulated break in this line is limited to the CVCS purification normal flow rate of 379 L/min (100 gpm). In DCD Tier 2, Section 15.6.2, considering the low temperature of the breakflow and the reduced iodine activity because of demineralization for the postulated break in the discharge line from the CVCS to the WLS, the applicant proposed, 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 breakflow to less than 492 L/min (130 gpm). Because the sample line isolation valves are open only 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 breakflow isolation time will be less than 30 minutes. The applicant assumed the fluid escaping the break 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 Section 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 SFP cooling capability. Section 15.3.9 of this report discusses the staff's review of the radiological consequences of SFP boiling.

The staff has reviewed 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 applicant's 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 operate at the AP1000 TS equilibrium concentration limit of 37 kBq/gm (1.0 µCi/gm) for 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 a rising iodine concentration in the primary coolant during the course of the accident.

Tables 15.3-5 and 15.3-1 of this report provide the major parameters and assumptions used by the staff and the results of the staff's radiological consequence analyses, respectively. 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 SFP cooling capability will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 50.34.

In DCD Tier 2, Section 6.4.4, Westinghouse reported the results of its radiological consequence analysis for personnel in the main control room during a design-basis small line break accident, relying on the VES to limit the radioactivity to which the personnel may be exposed. Section 6.4 of this report describes the staff's review and assessment of the VES. To verify the Westinghouse assessment, the staff performed an independent radiological consequence calculation for the small line break accident with VES operation under high-high radiation levels. The staff finds reasonable assurance that the VES, under high-high radiological conditions as described in DCD Tier 2, Section 6.4, can mitigate the dose in the main control room following a design-basis small line break accident to meet the dose criterion specified in GDC 19.
Tables 15.3-9 and 15.3-1 of this report provide the major assumptions used by the staff and the resulting radiological consequence analysis results for the control room operators, respectively.

The calculation of the dose to the control room operators does not rely on the nuclear island VBS, a non-safety-related system, to meet the requirements of GDC 19. Under some accident circumstances, the non-safety-related VBS would be available to pressurize the control room and the TSC with filtered air and to provide recirculation cleanup. Section 9.4 of this report describes the staff's review and assessment of the non-safety-related VBS. In DCD Tier 2, Section 6.4.4, Westinghouse also reported the results of its radiological consequence analysis for personnel in the main control room and the TSC during a design-basis small line break accident with the VBS available. The staff finds reasonable the Westinghouse assertion that, if available, the VBS can mitigate the dose in the main control room and the TSC following a design-basis small line break accident to be within 0.05 Sv (5 rem) TEDE. Table 15.3-9 of this report provides the major assumptions used by the staff and the resulting radiological consequence analysis results for the personnel in the control room and TSC with VBS available.

### 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 SFP cooling capability. Section 15.3.9 of this report discusses the staff's review of the radiological consequences of SFP boiling.

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 operate at the AP1000 TS equilibrium iodine concentration limit of 37 kBq/gm (1.0 µCi/gm) for DEI-131 in the primary coolant. The iodine spike generated during the accident is assumed to increase the release rate of iodine from the fuel by a factor of 335, resulting in a rising iodine concentration in the primary coolant during the course of the accident.

For Case 2, the 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) for DEI-131, specified in the AP1000 TS. Tables 15.3-6 and 15.3-1 of this report provide the major parameters and assumptions used by the staff and the results of the staff's radiological consequence analyses for the EAB and LPZ and for the control room, respectively. 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 SFP cooling capability will not exceed a small fraction (i.e., 10 percent or 0.025 Sv (2.5 rem) TEDE) of the dose criterion set forth in 10 CFR 50.34.

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) for DEI-131 and coincident with the loss of SFP cooling capability will not exceed the dose criterion set forth in 10 CFR 50.34 (i.e., 0.25 Sv (25 rem) TEDE).

In DCD Tier 2, Section 6.4.4, Westinghouse reported the results of its radiological consequence analysis for personnel in the main control room during a design-basis SGTR, relying on the VES to limit the radioactivity to which the personnel may be exposed. Section 6.4 of this report describes the staff's review and assessment of the VES. To verify the Westinghouse assessment, the staff performed an independent radiological consequence calculation for the

SGTR with VES operation under high-high radiation levels. The staff finds reasonable assurance that the VES, under high-high radiological conditions as described in DCD Tier 2, Section 6.4, can mitigate the dose in the main control room following a design-basis SGTR to meet the dose criterion specified in GDC 19. Tables 15.3-9 and 15.3-1 of this report provide the major assumptions used by the staff and the resulting radiological consequence analysis results for the control room operators, respectively.

The calculation of the dose to the control room operators does not rely on the nuclear island VBS, a non-safety-related system, to meet the requirements of GDC 19. Under some accident circumstances, the non-safety-related VBS would be available to pressurize the control room and the TSC with filtered air and to provide recirculation cleanup. Section 9.4 of this report describes the staff's review and assessment of the non-safety-related VBS. In DCD Tier 2, Section 6.4.4, Westinghouse also reported the results of its radiological consequence analysis for personnel in the main control room and the TSC during a design-basis SGTR with the VBS available. The staff finds reasonable the Westinghouse assertion that, if available, the VBS can mitigate the dose in the main control room and the TSC following a design-basis SGTR to be within 0.05 Sv (5 rem) TEDE. Table 15.3-9 of this report provides the major assumptions used by the staff and the resulting radiological consequence analysis results for the personnel in the control room and the TSC with VBS available.

### 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 concluded that certain bounding sets of atmospheric relative concentration values (i.e., $\chi$/Qs) specified in DCD Tier 2, Section 2.3, in conjunction with the use of natural deposition of fission product aerosol within the containment and the control of the pH of the water in the containment to prevent iodine evolution, are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated design-basis LOCA will fall within the relevant dose criteria established in 10 CFR 50.34 and in GDC 19.

All of the fission product releases stemming from the LOCA result from containment leakage. The AP1000 design does not have ESF systems outside containment. Therefore, the radiological consequence analyses do not consider leakage from the ESF systems. The containment was assumed to leak at its design leak rate of 0.1 wt% per day for the first 24 hours and at half of that rate (0.05 wt% per day) for the remaining 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 SFP cooling capability. Section 15.3.9 of this report discusses the staff's review of the radiological consequences of SFP boiling.

For the DSER, the staff had not completed its evaluation of the applicant's assumptions on aerosol removal in containment, as discussed in RAIs 470.009 and 470.011. Until the resolution of the issues with aerosol removal in containment, the staff could not complete its independent radiological consequence analysis of a postulated design-basis LOCA. This was Open Item 15.3.6-1. With the resolution of DSER Open Item 15.3-1 on aerosol removal, as discussed below, the staff could complete its independent analyses to confirm the applicant's assessment of the radiological consequences of the design-basis LOCA. Therefore, DSER Open Item 15.3.6-1 is closed.

The AP1000 design does not provide an active containment atmosphere cleanup system. Instead, the design relies on natural aerosol removal processes for deposition in the containment, such as gravitational settling and plateout on containment surfaces through diffusiophoresis and thermophoresis. Appendix 15B to DCD Tier 2 discusses the removal of airborne activity from the containment atmosphere. The applicant has provided a containment spray system for accident management following a severe accident as part of the AP1000 fire protection system design (discussed in 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, radiological consequence assessments give the containment spray system no credit for mitigation of radiological releases following a DBA.

The AP1000 LOCA offsite doses do not scale up from the AP600 proportionately with the change in the power or T-H conditions in the containment. The main reason is that, given the same leak rates, the amount of the release is a sole function of the airborne fission product mass, which is proportional to the core inventory. The AP1000 inventory is about twice that of the AP600 (about 70 percent higher power with a longer fuel cycle). Consequently, the removal rate for the AP1000 must be at least twice that of the AP600 to achieve the same offsite doses. The calculations performed by Westinghouse and the confirmatory uncertainty analysis performed by the staff show that the current containment design does not provide for a removal rate of twice that of the AP600. However, by adjusting the atmospheric dispersion factors ( $\chi$/Qs) lower than those used in the AP600 design, the radiological consequences of the LOCA remain acceptable. Because of this, the siting of the AP1000 design is more limited than for the AP600.

Applying credit for aerosol removal through natural processes requires input from T-H and aerosol behavior models. The basis document defining the revised accident source term, NUREG-1465, does not specify an associated T-H scenario, or methodology or acceptance criteria for aerosol removal. The alternative source term regulatory guidance, RG 1.183, also does not specify these items. NUREG-1465 describes a source term that was derived from an examination of a set of severe accident sequences for LWRs and is intended to be representative or typical and does not imply a specific scenario, much less the worst case.

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 PCS to inhibit stratification of the containment atmosphere. DCD Tier 2, Appendix 6A, discusses the physical mechanisms of natural circulation mixing that occur in the AP1000. Section 6.2.5.3 of this report provides the staff's discussion of natural circulation within the AP1000 containment.

In DCD Tier 2, Table 15B-1, the applicant provided aerosol removal coefficient values starting at the onset of a gap release through the first 24 hours into a DBA. The values range from 0.287/h to 1.141/h. The aerosol removal coefficients calculated by the applicant neglect steam condensation on the airborne particles, turbulent diffusion, and turbulent agglomeration. The assumed source aerosol size is conservatively small, because it is at the low end of the mass mean aerosol size range of 1.5 µm to 5.5 µm used in NUREG/CR-5966, “A Simplified Model of Aerosol Removal by Containment Sprays,” issued June 1993. Selection of a small aerosol size would underestimate removal by sedimentation.

The aerosol removal coefficients calculated by Westinghouse for the AP1000 are based on a single set of parameters, and Westinghouse performed sensitivity studies to determine if changing the parameters would affect the overall removal coefficients. The staff felt that some parameters chosen originally by Westinghouse were not conservative, as would be required for use in design-basis analyses, nor were these parameters likely to be suitable for use as point estimates, considering the lack of knowledge and uncertainty in some areas of post-LOCA aerosol formation and detailed containment response.

In the past, the staff and industry evaluated aerosol removal through well-established models of spray removal or condensation. The AP1000 application relies on natural deposition processes that depend strongly on local T-H conditions. While gravitational settling is relatively easy to understand, aerosol removal through diffusiophoresis and thermophoresis is much more complex. Diffusiophoresis is associated with steam condensation on the heat sinks and depends on the condensation steam mass flux. Thermophoresis relies only on the temperature gradient close to the surface on which the particles would be deposited. Thermophoresis is more subtle than the other two natural deposition processes. Because the temperature gradient cannot be measured or easily calculated, its model uses the heat flux at the surface divided by the thermal conductivity of the gas adjacent to the surface as an equivalent measure of the driving force. Simultaneous occurrence of the two phoretic processes introduces an additional level of complexity.

The Westinghouse methodology includes industry’s Modular Accident Analysis Program (MAAP) code, an integrated accident analysis program, to establish T-H boundary conditions as an input to an aerosol code (STARNAUA). To determine the acceptability of the Westinghouse modeling, the staff audited Westinghouse calculations of the containment removal coefficients. The audit revealed that the heat flux used by Westinghouse included the convection, the thermal radiation, and the decay heat from airborne fission products. Thermal radiation and decay heat do not contribute to the temperature gradient that drives thermophoresis and their use caused the overall aerosol removal to be nonrealistic and nonconservative. Westinghouse recalculated the overall aerosol removal coefficients by correcting this error.

In its independent evaluation of aerosol removal coefficients, the staff considered the same natural processes for removing aerosols from the containment atmosphere over the entire period of an accident (30 days). These processes include the sedimentation mechanism of gravitational settling, such as aerosol agglomeration, and the phoretic mechanisms of diffusiophoresis and thermophoresis.

The staff contracted Sandia National Laboratory to perform quantitative analyses of uncertainties in predicting the aerosol removal rates. The initiating event, used in the analysis (3BE-1 sequence), is a double-ended break of a DVI 10.2-cm- (4-in.-) diameter line. The break is assumed to be in a larger compartment of the passive core cooling system (PXS). The sequence includes the water spillage into the PXS-B compartment from one of the lost accumulators and one of the CMTs. The analysis assumes that one CMT and one accumulator (in the intact DVI train) remain available to inject water into the RV, which is not sufficient to keep the core covered. This deficiency in core cooling leads to degradation and/or melt of the fuel. Eventually, the water level in the containment reaches the break level and spills into the vessel, arresting the core degradation before potential failure of the vessel lower head.

As part of the staff's review, the aerosol behavior in the AP1000 containment was predicted using the MELCOR integrated accident analysis code, which includes the MAEROS aerosol mechanics code. The staff used the results of a fully integrated MELCOR analysis of the AP1000 3BE-1 accident, performed by the staff's contractor, Energy Research, Inc., to develop a simplified containment model for the uncertainty analysis. The NUREG-1465 radiological source term for the gap release and in-vessel release phases were used in place of the source term predicted in the fully integrated analysis. The uncertainty analysis considered those MELCOR parameters known to affect aerosol settling and depletion to be uncertain within a range of values, represented by an assumed distribution function. A Monte Carlo method, which randomly samples the uncertain parameters and performs a large number of separate MELCOR analyses, estimated the effect of the uncertain physics parameters on the aerosol removal rate.

In its evaluation of aerosol removal rates, the staff used the containment geometry (e.g., volume, upward facing surface area) provided by Westinghouse and the fission product release timing, fractions, and release rates as described in NUREG-1465.

The principal uncertainties in aerosol properties and aerosol behavior considered in the staff's analyses included the following:

- aerosol size and distribution
- aerosol void fraction and particle shape factors
- aerosol material density
- nonradioactive aerosol mass
- particle slip coefficient
- sticking probability for agglomeration
- boundary layer thickness for diffusion deposition
- thermal accommodation coefficient for thermophoresis
- ratio of thermal conductivity of particle to gas
- turbulent energy dissipation
- multipliers on heat and mass transfer to containment shell

The Westinghouse calculation of aerosol removal coefficients is based on an analysis of a single T-H scenario and uses a single aerosol model without providing an uncertainty analysis. The staff believes that the Westinghouse approach, though potentially acceptable, represents a single BE result. Westinghouse used T-H conditions associated with the 3BE-1 severe accident sequence. The staff concludes that using the T-H conditions associated with the 3BE-1 severe accident sequence represents the spectrum of accidents evaluated for the AP1000 for the following reasons:

- The conditions are representative of the 3BE accident class, which is the dominant contributor to the core damage frequency for the AP1000.

- The T-H conditions for 3BE accidents are typical of most of the analyzed sequences because the majority of severe accident sequences analyzed for the AP1000 design are fully depressurized and reflooded, given the highly reliable ADS.

- The corresponding T-H profiles for these depressurized and reflooded cases are sufficiently similar.

- The use of a fully depressurized, low-pressure accident sequence in conjunction with the source term described in NUREG-1465 is appropriate because the release fractions for the source terms presented in NUREG-1465 are intended to be representative or typical of those associated with a low-pressure core melt accident.

Therefore, the staff concludes that the 3BE-1 accident sequence is appropriate for determining the amount of credit to give to the natural aerosol removal processes in the AP1000 containment. In its independent uncertainty analysis, the staff did not employ the T-H calculated by Westinghouse using the MAAP code. The staff's uncertainty analysis did not include differences between the staff and Westinghouse calculations with respect to containment T-H and containment modeling as variables for study.

Although the choice of scenario is acceptable, the staff believes that a BE approach requires an evaluation of the associated uncertainties. The staff used an alternative T-H code (MELCOR) as an input to a Monte Carlo sampling (120 runs) of the above-listed parameters affecting aerosol behavior. Engineering judgment was used to choose parameters as well as for the range and distribution of their values, after several discussions between the staff and the contractor. The resultant distribution of possible aerosol removal coefficients has a 95-percent level of confidence. The 5$^{th}$, 20$^{th}$, 40$^{th}$, 45$^{th}$, 50$^{th}$, 55$^{th}$, 60$^{th}$, 80$^{th}$, and 95$^{th}$ percentiles, as depicted in Figure 15.3.6-1 of this report and documented in Table 15.3.6-1 of this report, provide the uncertainty distribution.

In the uncertainty analysis performed for the staff, the conservative lower bound (5$^{th}$ percentile) aerosol removal coefficient ranges from 0.07/h to 1.26/h for the first 24 hours into a DBA (shown in Table 15.3-8 of this report). The BE median (50$^{th}$ percentile) aerosol removal coefficient ranges from 0.3/h to 1.35/h. The traditional regulatory approach is to accept a bounding value, which would generally be the 5$^{th}$ percentile, as the maximum bounding value, where there is an estimated 5 percent chance that the aerosol removal coefficients will be lower than assumed, thereby resulting in higher calculated doses. In this particular case, however, the staff proposes the use of the median value for the following reasons:

- In an alternative source term (AST) pilot application for Perry, the staff had previously accepted the 50$^{th}$ percentile value for steamline deposition, based on the NRC Office of Nuclear Regulatory Research opinion that it is appropriate given other conservatisms built into the other parts of the analysis.

- The staff believes that the selected scenario belongs to a worst-case category.

- The median values are least affected by the user's sampling initial conditions.

- The choice of the initial ranges and distributions of the selected parameters is highly subjective.
- Both the AST and the modeling of containment leakage have built-in conservativisms.
- The dose calculation code requires yet another averaging of the aerosol removal coefficients for the specified time periods.
- The independent MELCOR-calculated aerosol removal coefficients are grouped mostly well above the 5$^{th}$ percentile lower bound.

It is important to emphasize that it is the staff’s judgment that the acceptance of the 50$^{th}$ percentile for use in DBA analyses is appropriate for this particular safety analysis because an underlying conservative bias has been built into the staff’s uncertainty methodology (e.g., the choice of the ranges and distributions of the sampled parameters). The staff believes that different choices of initial ranges and distributions, and/or the use of a different uncertainty methodology, may not be acceptable.

The uncertainty analysis also indicated that, for the staff’s model as implemented in MELCOR, phoretic processes (thermophoresis and diffusiophoresis) together dominate for the first 4 hours of the LOCA and constitute 90 percent of aerosol removal. After that, gravitational settling is the dominant removal mechanism.

Although the Westinghouse-calculated aerosol removal coefficient values lie within the staff’s uncertainty analysis upper and lower bounds for portions of the early part of the accident, they generally do not lie below the median value calculated by the staff. While the staff does find that gravitational settling, thermophoresis, and diffusiophoresis are physical processes that occur in the AP1000 containment, and credit may be taken for aerosol removal through these processes, the staff does not approve the specific Westinghouse-calculated aerosol removal coefficient values. These values are an intermediate product used in the dose analysis and are not subject to any regulations, per se. Staff independent analyses using the Westinghouse-supplied $\chi$/Qs and plant parameters result in acceptable doses, as discussed below. Thus, while the staff and Westinghouse diverge on values for the intermediate steps in the dose calculations, the staff agrees with the overall conclusion that the AP1000 design results in acceptable doses.

The staff performed an independent dose analysis with the median aerosol removal coefficient values from the staff’s uncertainty analysis, along with other analysis parameters and the bounding hypothetical atmospheric dispersion factors provided by Westinghouse, and the results fall within the dose criteria of 10 CFR 50.34 and GDC 19. The staff performed a sensitivity analysis, which resulted in calculated doses that remain below the regulatory acceptance criteria for aerosol removal coefficients as low as those given in the 20$^{th}$ percentile in the staff’s uncertainty analysis.

The staff finds the radiological consequence analysis of the postulated DBA LOCA acceptable, based on the Westinghouse DCD Tier 2, Chapter 15, plant parameters used in the staff’s analysis, the staff-calculated aerosol removal coefficient estimates (50$^{th}$ percentile, 95 percent confidence), and the latest revision to the AP1000 $\chi$/Qs, as documented in the Revision 5 response to DSER Open Item 15.3-1, dated June 21, 2004. Westinghouse will include the revised information, including $\chi$/Qs, in revision 12 of the AP1000 DCD and the staff will confirm. With this basis, the doses meet the regulatory criteria of 10 CFR 50.34 and GDC 19 and are, therefore, acceptable.

In the DSER, the staff stated that it would perform an independent 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 was identified as DSER Open Item 15.3-1. The staff has completed its evaluation and finds that the differences between the resulting radiological consequences calculated using either Westinghouse's or the staff's aerosol removal rates are insignificant. Accordingly, analysis credit for aerosol removal through natural processes is found acceptable, and DSER Open Item 15.3-1 is closed.

Because of the unique nature of the AP1000 design, which enhances natural aerosol removal phenomena (such as the enhanced condensation of steam by external cooling of the containment vessel instead of an internal containment spray), the staff has approved the use of this T-H profile specifically for the AP1000. The NRC does not intend credit for aerosol removal because of diffusiophoresis and thermophoresis to be generic for other plant designs, and this practice must be approved on a case-by-case basis. Because the staff did not explicitly find the input values for the aerosol removal coefficient used in the Westinghouse dose calculation to be acceptable, the NRC staff must review any changes to the LOCA calculation made by a COL applicant or licensee of an AP1000 plant.

In DCD Tier 2, Section 6.4.4, Westinghouse reported the results of its radiological consequence analysis for personnel in the main control room during a design-basis LOCA, relying on the VES to limit the radioactivity to which the personnel may be exposed. Section 6.4 of this report describes the staff's review and assessment of the VES. To verify the Westinghouse assessment, the staff performed an independent radiological consequence calculation for the LOCA with VES operation under high-high radiation levels. The staff finds reasonable assurance that the VES, under high-high radiological conditions as described in DCD Tier 2, Section 6.4, can mitigate the dose in the main control room following a design-basis LOCA to meet the dose criterion specified in GDC 19. Tables 15.3-9 and 15.3-1 of this report provide the major assumptions used by the staff and the resulting radiological consequence analysis results for the control room operators, respectively.

The calculation of the dose to the control room operators does not rely on the nuclear island VBS, a non-safety-related system, to meet the requirements of GDC 19. Under some accident circumstances, the non-safety-related VBS would be available to pressurize the control room and the TSC with filtered air and to provide recirculation cleanup. Section 9.4 of this report describes the staff's review and assessment of the non-safety-related VBS. In DCD Tier 2, Section 6.4.4, Westinghouse also reported the results of its radiological consequence analysis for personnel in the main control room and the TSC during a design-basis LOCA with the VBS available. The staff finds the Westinghouse assertion reasonable that, if available, the VBS can mitigate the dose in the main control room and the TSC following a design-basis LOCA to be within 0.05 Sv (5 rem) TEDE. Table 15.3-9 of this report provides the major assumptions used by the staff and the resulting confirmatory analysis results.

### 15.3.7 Radiological Consequences of a Fuel-Handling Accident

In DCD Tier 2, Section 15.7.4, the applicant presented its analyses of the radiological consequences of a postulated 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, closure of the containment purge lines based on the detection of high airborne radioactivity can terminate the release of fission products. 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 that a single fuel assembly that has undergone 24 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. The fuel pool water has an assumed effective decontamination factor of 200 for total iodine. These gap fractions agree with RG 1.183 guidance. The applicant assumed that iodine in the particulate form is not volatile and, therefore, is not released. In accordance with RG 1.183 guidance, the applicant assumed that the particulate CsI is converted instantaneously to the elemental form of iodine when it is released from the fuel into the low-pH pool water.

The applicant also considered a coincident loss of SFP cooling capability. Section 15.3.9 of this report discusses the staff's review of the radiological consequences of SFP boiling.

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 FHA occurring 24 hours after shutdown, coincident with a loss of the SFP cooling capability. Tables 15.3-10 and 15.3-1 of this report provide the major parameters and assumptions used by the staff and the results of the staff's radiological consequence analyses, respectively. 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 FHA at 24 hours after shutdown with the loss of SFP cooling capability will fall well within the dose criterion set forth in 10 CFR 50.34 (i.e., 25 percent or 0.063 Sv (6.3 rem) TEDE).

In RAI 630.052, the NRC staff asked the applicant to justify not including 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 it 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 the DCD to include a paragraph discussing the evaluation of the FHA, assuming a decay time of 24 hours. The applicant asserted that a decay time LCO is not necessary 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 believes that, in order to exclude a TS 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 continued to include the 100-hour decay time assumption. Additionally, in the DCD Tier 2, Section 15.7.4.5, discussion of the evaluation of the FHA at 24 hours, the applicant did not discuss the impact of the decay time on control room habitability from an FHA at 24 hours. The staff did not consider this issue to be resolved at the time of issuance of the DSER. This was Open Item 15.3.7-1. To resolve this issue, the applicant revised the design-basis FHA to assume that the fuel breached has been part of a critical operating core 24 hours before the accident occurs. The revised analysis evaluated the radiological consequences offsite and in the control room. The applicant also added refueling operations TS 3.9.7, “Decay Time,” which requires that the reactor be shut down for at least 100 hours before movement of irradiated fuel in the reactor pressure vessel. Therefore, DSER Open Item 15.3.7-1 is closed.

In DCD Tier 2, Section 6.4.4, Westinghouse reported the results of its radiological consequence analysis for personnel in the main control room during a design-basis FHA, relying on the VES to limit the radioactivity to which the personnel may be exposed. Section 6.4 of this report describes the staff’s review and assessment of the VES. To verify the Westinghouse assessment, the staff performed an independent radiological consequence calculation for the FHA with VES operation under high-high radiation levels. The staff finds reasonable assurance that the VES, under high-high radiological conditions as described in DCD Tier 2, Section 6.4, can mitigate the dose in the main control room following a design-basis FHA to meet the dose criterion specified in GDC 19. Tables 15.3-9 and 15.3-1 of this report provide the major assumptions used by the staff and the resulting radiological consequence analysis results for the control room operators, respectively.

The calculation of the dose to the control room operators does not rely on the nuclear island VBS, a non-safety-related component, to meet the requirements of GDC 19. Under some accident circumstances, the non-safety-related VBS would be available to pressurize the control room and the TSC with filtered air and to provide recirculation cleanup. Section 9.4 of this report describes the staff’s review and assessment of the non-safety-related VBS. In DCD Tier 2, Section 6.4.4, Westinghouse also reported the results of its radiological consequence analysis for personnel in the main control room and the TSC during a design-basis FHA with the VBS available. The staff finds reasonable the Westinghouse assertion that, if available, the VBS can mitigate the dose in the main control room and the TSC following a design-basis FHA to be within 0.05 Sv (5 rem) TEDE. Table 15.3-9 of this report provides the major assumptions used by the staff and the resulting radiological consequence analysis results for the personnel in the control room and the TSC with VBS available.

### 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 tank containing liquid. The following regulations provided the basis for the acceptance criteria specified in this SRP section:

- GDC 60, "Control of Releases of Radioactive Materials to the Environment," as it relates to the ability of the design of the radioactive waste management system to control the 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) 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). The design of the WLS incorporates specific design features to mitigate the effects of failure, if the WLS 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 floor drains would drain the liquid 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 applicant 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 waterbodies 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 DBA, Westinghouse evaluated the added radiological consequences of SFP boiling because of the loss of SFP cooling capability.

The SFS is designed to perform the following functions:

- remove heat from the SFP and the IRWST
- remove radioactive corrosion and fission products from the SFP, the IRWST, and the 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, an HX, a filter, and a demineralizer. However, the SFS 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 are 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's assessment, the staff performed an independent radiological consequence calculation for the loss of the SFP cooling capability. Tables 15.3-11 and 15.3-1 of this report provide the major parameters and assumptions used by the staff and the results of the staff's radiological consequence analyses, respectively. 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 DBAs described in DCD Tier 2, Chapter 15, for the AP1000 design. Based on the evaluation discussed above, the staff concludes that the AP1000 design meets 10 CFR 50.34(a)(1)(ii)(D) dose criteria and the offsite dose acceptance criteria, as given in RG 1.183 for these accidents.

The staff finds reasonable assurance that the VES, under high-high radiological conditions as described in DCD Tier 2, Section 6.4, can mitigate the dose in the main control room following DBAs to meet the dose criterion specified in GDC 19.

The staff finds it reasonable that, if available, the non-safety-related VBS can mitigate the dose in the main control room and the TSC following DBAs to be within 0.05 Sv (5 rem) TEDE.

COL Action Item 15.3.8-1, regarding a site-specific analysis of the offsite radiological consequences of a liquid tank failure, explains that 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.

Fig. 15.2.7.6

Pressure Drop vs Core Steaming Rate

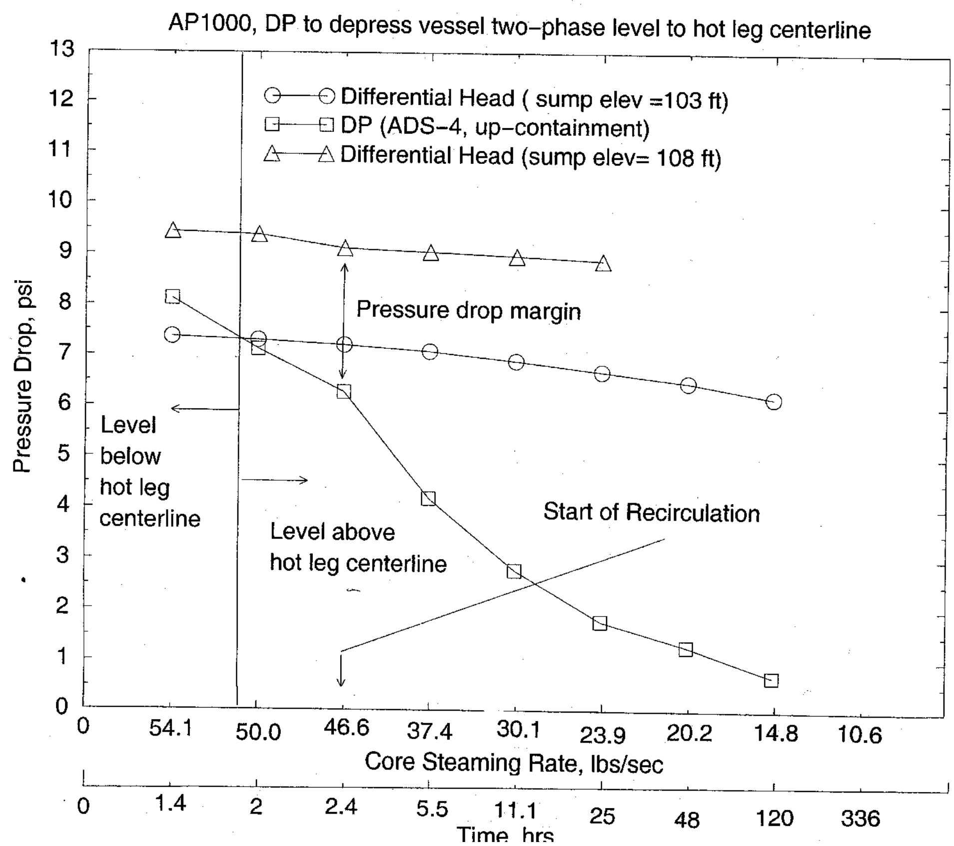

Table 15.3-1 Staff-Calculated Radiological Consequences of Design-Basis Accidents (Total Effective Dose Equivalent (TEDE))

| Postulated Accident | EAB | LPZ | Control Room |
|---|---|---|---|
| Loss of coolant accident | 190 mSv (19 rem) | 150 mSv (15 rem) | 34 mSv (3.4 rem) |
| Main steamline break outside containment | | | |
| With accident-initiated iodine spike | 2 mSv (0.2 rem) | 8 mSv (0.8 rem) | 13 mSv (1.3 rem) |
| With preaccident iodine spike | <1 mSv (<0.1 rem) | 1 mSv (0.1 rem) | 9 mSv (0.9 rem) |
| Reactor coolant pump shaft seizure | | | |
| With feedwater available | <1 mSv (<0.1 rem) | <1 mSv (<0.1 rem) | 8 mSv (0.8 rem) |
| Without feedwater available | <1 mSv (<0.1 rem) | <1 mSv (<0.1 rem) | 12 mSv (1.2 rem) |
| Rod ejection accident | 15 mSv (1.5 rem) | 24 mSv (2.4 rem) | 11 mSv (1.1 rem) |
| Fuel-handling accident | 24 mSv (2.4 rem) | 10 mSv (1.0 rem) | 29 mSv (2.9 rem) |
| Small line break accident | 10 mSv (1.0 rem) | 4 mSv (0.4 rem) | 14 mSv (1.4 rem) |
| Steam generator tube rupture | | | |
| With accident-initiated iodine spike | 5 mSv (0.5 rem) | 7 mSv (0.7 rem) | 26 mSv (2.6 rem) |
| With preaccident iodine spike | 10 mSv (1.0 rem) | 6 mSv (0.6 rem) | 50 mSv (5 rem) |
| Spent fuel pool boiling | n/a* | <0.1 mSv (<0.01 rem) | <0.1 mSv (<0.01 rem) |

* n/a, not applicable

Table 15.3-2 Assumptions Used to Evaluate the Radiological Consequences of the Main Steamline Break Accident Outside Containment

| Parameter | Value | |
|---|---|---|
| Power level, MWt | 3468 | |
| Reactor primary coolant iodine concentrations | | |
| Accident-initiated iodine spike, KBq/gm DEI-131 | 37 | (1.0 µCi/gm) |
| Preaccident iodine spike, MBq/gm DEI-131 | 2.2 | (60 µCi/gm) |
| Steam generator in faulted loop | | |
| Initial water mass, kg | 1.37E+5 | (3.03E5 lb) |
| Primary to secondary leak rate, L/d | 567.8 | (150 gpd) |
| Iodine partition coefficient | | 1.0 |
| Steam generator in intact loop | | |
| Primary to secondary leak rate, L/d | 567.8 | (150 gpd) |
| Iodine partition coefficient | | 1.0 |
| Steam released, kg | | |
| 0 to 2 hr | 1.37E+5 | (3.0335E5 lb) |
| 2 to 8 hr | 5.56E+3 | (1.225E4 lb) |
| Ratio of iodine release rate from fuel during iodine spike to that during steady-state operation | 500 | |
| Reactor primary coolant mass, kg | 1.74E+5 | (3.84E+5 lb) |
| Duration of accident, hr | 72 | |
| Atmospheric dispersion values, sec/m$^3$ | | |
| EAB | | |
| 0 to 2 hours | 5.1E-4 | |
| LPZ | | |
| 0 to 8 hours | 2.2E-4 | |
| 8 to 24 hours | 1.6E-4 | |
| 1 to 4 days | 1.0E-4 | |
| 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 | 0.10 | |
| 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, MBq/gm DEI-131 | 2.2 | (60 µCi/gm) |
| Secondary coolant mass, kg | 2.75E+5 | (6.06E5 lb) |
| Primary to secondary leak rate, kg/hr | 47.3 | (104.3 lb/hr) |
| Iodine partition coefficient | 0.01 | |
| Alkali metal partition coefficient | 0.001 | |
| Steam released, kg | | |
| 0 to 1.5 hr | 2.94E+5 | (6.48E+5 lb) |
| Leak flashing fraction | | |
| 0 to 60 min | 0.04 | |
| >60 min | 0 | |
| Reactor primary coolant mass, kg | 1.68E+5 | (3.7E+5 lb) |
| Duration of accident, hr | 1.5 | |
| Atmospheric dispersion values, sec/$m^3$ | | |
| EAB | | |
| 0 to 2 hours | 5.1E-4 | |
| LPZ | | |
| 0 to 8 hours | 2.2E-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, MBq/gm DEI-131 | 2.2 | (60 µCi/gm) |
| Reactor coolant mass, kg | 1.68E+5 | (3.7E5 lb) |
| 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, kg/hr | 47.31 | (104.3 lb/hr) |
| Leak flashing fraction | 0.04 | |
| Secondary coolant mass, kg | 2.75E+5 | (6.06E+5 lb) |
| Duration of steam release from secondary system, sec | 1800 | |
| Steam released from secondary system, kg | 4.9E+4 | (1.08E+5 lb) |
| 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 to 24 hr | 0.10 |
| >24 hr | 0.05 |
| 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 | 5.1E-4 |
| LPZ | |
| 0 to 8 hours | 2.2E-4 |
| 8 to 24 hours | 1.6E-4 |
| 1 to 4 days | 1.0E-4 |
| 4 to 30 days | 8.0E-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, KBq/gm DEI-131 | 37 | (1.0 µCi/gm) |
| Preaccident iodine spike, MBq/gm DEI-131 | 2.2 | (60 µCi/gm) |
| Ratio of iodine release rate from fuel during iodine spike to that during steady-state operation | 500 | |
| Reactor coolant mass, kg | 1.68E+5 | (3.7E+5 lb) |
| Duration of accident, min | 30 | |
| Sample line break flow, L/m | 492.1 | (130 gpm) |
| Fraction of reactor coolant flashing | 0.41 | |
| Atmospheric dispersion values, sec/m$^3$ | | |
| EAB | | |
| 0 to 2 hours | 5.1E-4 | |
| LPZ | | |
| 0 to 8 hours | 2.2E-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, KBq/gm DEI-131 | 37 | (1.0 µCi/gm) |
| Preaccident iodine spike, MBq/gm DEI-131 | 2.2 | (60 µCi/gm) |
| Steam generator in faulted loop | | |
| Initial water mass, kg | 1.37E+5 | (3.03E5 lb) |
| Primary to secondary leak rate, L/d | 567.8 | (150 gpd) |
| Iodine partition coefficient for breakflow | 1 | |
| Iodine partition coefficient for secondary steaming | 0.01 | |
| Alkali metal partition coefficient | 0.001 | |
| Steam generator in intact loop | | |
| Primary to secondary leak rate, L/d | 567.8 | (150 gpd) |
| Iodine partition coefficient | 0.01 | |
| Alkali metal partition coefficient | 0.001 | |
| Steam released, kg | | |
| 0 to 2 hr | 1.65E+5 | (3.64E+5 lb) |
| 2 to 8 hr | 3.24E+5 | (7.15E+5 lb) |
| Ratio of iodine release rate from fuel during iodine spike to that during steady-state operation | 335 | |
| Reactor primary coolant mass, kg | 1.74E+5 | (3.84E+5 lb) |
| Duration of accident, hr | 13.19 | |
| Atmospheric dispersion values, sec/$m^3$ | | |
| EAB | | |
| 0 to 2 hours | 5.1E-4 | |
| LPZ | | |
| 0 to 8 hours | 2.2E-4 | |
| 8 to 24 hours | 1.6E-4 | |
| 1 to 4 days | 1.0E-4 | |
| 4 to 30 days | 8.0E-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 | | |
| | Gap Release | In-vessel Release |
| Nuclide Group | (0–0.5 hr) | (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 to 24 hours | | 0.1 |
| >24 hours | | 0.05 |
| Primary containment free volume, $m^3$ | | 5.83E+4 (2.06E6 $ft^3$) |
| Elemental iodine deposition removal coefficient, $hr^{-1}$ | | 1.7 |
| Decontamination factor limit for elemental iodine removal | | 200 |
| Removal coefficients for particulates | | Table 15.3-8 |
| Accident duration, days | | 30 |
| Atmospheric dispersion values, $sec/m^3$ | | |
| EAB | | |
| 0 to 2 hours | | 5.1E-4 |
| LPZ | | |
| 0 to 8 hours | | 2.2E-4 |
| 8 to 24 hours | | 1.6E-4 |
| 1 to 4 days | | 1.0E-4 |
| 4 to 30 days | | 8.0E-5 |
| Control room analysis parameters | | Table 15.3.9 |

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

| Time Post Release (hours) | Removal Rates ($hour^{-1}$) |
|---|---|
| 0.00 to 0.37 | 0.945 |
| 0.37 to 0.87 | 0.540 |
| 0.87 to 1.37 | 0.430 |
| 1.37 to 1.87 | 0.600 |
| 1.87 to 2.37 | 0.855 |
| 2.37 to 2.87 | 0.585 |
| 2.87 to 3.37 | 0.575 |
| 3.37 to 6.87 | 0.480 |
| 6.87 to 24.00 | 0.430 |

Table 15.3-9 (Sheet 1 of 2) Assumptions Used to Evaluate the Radiological Consequences to Control Room Operators Following a Design-Basis Accident

| Parameter | Value | |
|---|---|---|
| Accident release modeling | | |
| Main steamline 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 | |
| Loss-of-coolant accident | Table 15.3-7 | |
| Fuel-handling accident | Table 15.3-10 | |
| Control room free volume, $m^3$ | 1.01E+3 | (3.57E+4 $ft^3$) |
| 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 | 1.0 | |
| 24 to 96 hr | 0.6 | |
| 96 to 720 hr | 0.4 | |
| <u>Control room with emergency habitability system credited</u> | | |
| Initial interval before actuation of emergency habitability system | | |
| Air intake flow, $m^3$/min | 54.5 | (1925 cfm) |
| Intake filter efficiencies | N/A | |
| Interval with operation of emergency habitability system | | |
| Activity level at which emergency habitability system is actuated, KBg/$m^3$ of dose equivalent I-131 | 74 | (2.0E-6 Ci/$m^3$) |
| Flow from compressed air bottles, $m^3$/min | 1.7 | (60 cfm) |
| Unfiltered inleakage, $m^3$/min | 0.14 | (5 cfm) |
| Bottled air depletion time, hr | 72 | |
| Interval after depletion of bottled air supply | | |
| Air intake flow, $m^3$/min | 48.1 | (1700 cfm) |
| Intake filter efficiencies | N/A | |
| Recirculation flow | N/A | |

Table 15.3-9 (Sheet 2 of 2) Assumptions Used to Evaluate the Radiological Consequences to Control Room Operators Following a Design-Basis Accident

| Parameter | Value | |
|---|---|---|
| Time when compressed air restored, hr | 168 | |
| Staff-calculated DBA control room dose results | Table 15.3-1 | |
| <u>Control room and TSC with credit for supplemental air filtration mode of HVAC</u> | | |
| Initial interval before actuation of supplemental air filtration | | |
| Air intake flow, $m^3$/min | 54.5 | (1925 cfm) |
| Intake filter efficiencies | N/A | |
| Time delay to switch from normal operation to filtration, sec | 30 | |
| Filtered air intake flow, $m^3$/min | 24.4 | (860 cfm) |
| Filtered air recirculation flow, $m^3$/min | 77.6 | (2740 cfm) |
| Filter efficiency, % | | |
| Elemental iodine | 90 | |
| Organic iodine | 90 | |
| Particulates | 99 | |
| Unfiltered air inleakage, $m^3$/min | 2.55 | (90 cfm) |
| Staff-calculated TEDE in control room and TSC with credit for supplemental air filtration | | |
| Main steamline break | | |
| Accident-initiated I spiking | 28 mSv | (2.8 rem) |
| Preexisting I spike | 4 mSv | (0.4 rem) |
| Locked rotor | | |
| With feedwater available | 4 mSv | (0.4 rem) |
| Without feedwater available | <1 mSv | (<0.1 rem) |
| Rod ejection | 5 mSv | (0.5 rem) |
| Small line break | 4 mSv | (0.4 rem) |
| Steam generator tube rupture | | |
| Accident-initiated I spiking | 27 mSv | (2.7 rem) |
| Preexisting I spike | 29 mSv | (2.9 rem) |
| Loss-of-coolant accident | 32 mSv | (3.2 rem) |
| Fuel-handling accident | 14 mSv | (1.4 rem) |

Table 15.3-9a Atmospheric Dispersion Factors ( χ/Q) for Control Room Habitability Accident Dose Analysis

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

| | Plant Vent or PCS Air Diffuser | Ground Level Containment Release | PORV and Safety-Valve Releases | Steamline Break Releases | Fuel-Handling Area |
|---|---|---|---|---|---|
| 0–2 hr | 2.2E-3 | 2.2E-3 | 2.0E-2 | 2.4E-2 | 6.0E-3 |
| 2–8 hr | 1.4E-3 | 1.4E-3 | 1.8E-2 | 2.0E-2 | 4.0E-3 |
| 8–24 hr | 6.0E-4 | 6.0E-4 | 7.0E-3 | 7.5E-3 | 2.0E-3 |
| 24–96 hr | 4.5E-4 | 4.5E-4 | 5.0E-3 | 5.5E-3 | 1.5E-3 |
| 96–720 hr | 3.6E-4 | 3.6E-4 | 4.5E-3 | 5.0E-3 | 1.0E-3 |

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

| | Plant Vent or PCS Air Diffuser | Ground Level Containment Release | PORV and Safety-Valve Releases | Steamline Break Releases | Fuel-Handling Area |
|---|---|---|---|---|---|
| 0–2 hr | 6.6E-4 | 6.6E-4 | 4.0E-3 | 4.0E-3 | 6.0E-3 |
| 2–8 hr | 4.8E-4 | 4.8E-4 | 3.2E-3 | 3.2E-3 | 4.0E-3 |
| 8–24 hr | 2.1E-4 | 2.1E-4 | 1.2E-3 | 1.2E-3 | 2.0E-3 |
| 24–96 hr | 1.5E-4 | 1.5E-4 | 1.0E-3 | 1.0E-3 | 1.5E-3 |
| 96–720 hr | 1.3E-4 | 1.3E-4 | 8.0E-4 | 8.0E-4 | 1.0E-3 |

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 | 24 |
| 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/$m^3$ | |
| EAB | |
| 0 to 2 hours | 5.1E-4 |
| LPZ | |
| 0 to 8 hours | 2.2E-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, Bq | | |
| I-131 | 1.18E+11 | (3.18 Ci) |
| 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, Bq | 7.18E+10 | (1.94 Ci) |
| Initial pool water temperature, °C | 49 | (120 °F) |
| Time to initiate pool boiling, hr | 8.8 | |
| Steaming rate, kg/hr | | |
| 8.8–24 hr | 7348 | (16,200 lb/hr) |
| 24–48 hr | 7257 | (16,000 lb/hr) |
| 48–72 hr | 7121 | (15,700 lb/hr) |
| 72–88 hr | 6994 | (15,420 lb/hr) |
| 88–120 hr | 6917 | (15,250 lb/hr) |
| 120–168 hr | 6772 | (14,930 lb/hr) |
| ≥168 hr | 6577 | (14,500 lb/hr) |
| Iodine partition coefficient | 0.01 | |
| Atmospheric dispersion values, sec/$m^3$ | | |
| LPZ | | |
| 8 to 24 hours | 1.6E-4 | |
| 1 to 4 days | 1.0E-4 | |
| 4 to 30 days | 8.0E-5 | |
| Offsite breathing rate, $m^3$/sec | | |
| 8 to 24 hr | 1.8E-4 | |
| >24 hr | 2.3E-4 | |
| Control room analysis parameters | Table 15.3.9 | |

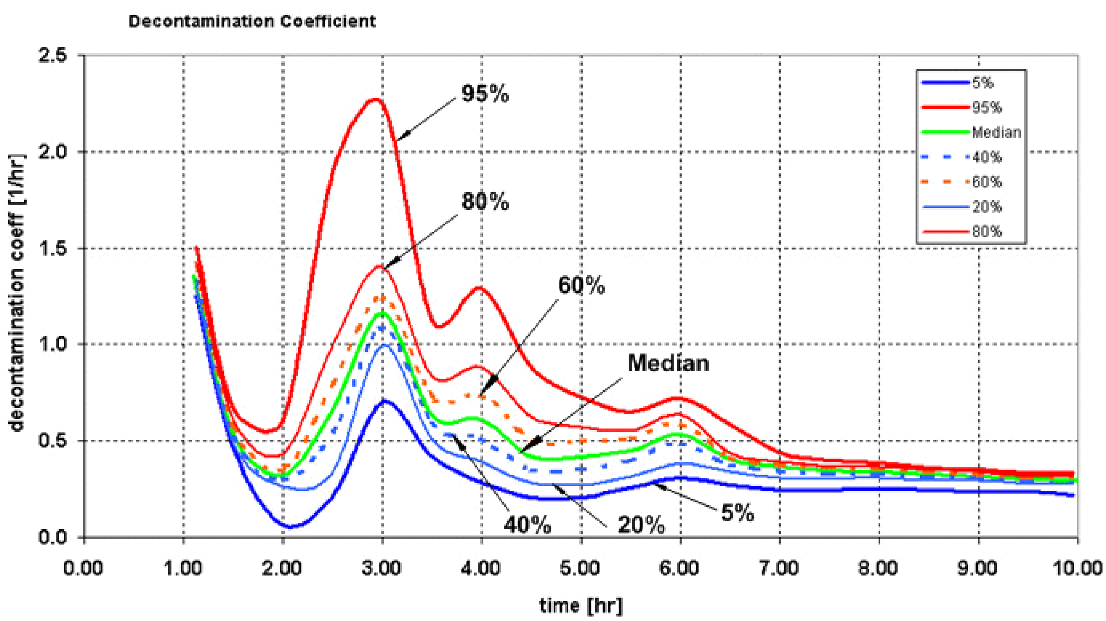


**Figure 15.3.6-1 Uncertainty Bands for Aerosol Removal Coefficients**

| Time hr | 5% | 95% | 40% | 60% | 45% | 55% | 20% | 80% | mean | median |
|---|---|---|---|---|---|---|---|---|---|---|
| 1.13 | 1.26 | 1.51 | 1.32 | 1.40 | 1.33 | 1.38 | 1.30 | 1.43 | 1.36 | 1.35 |
| 1.50 | 0.46 | 0.67 | 0.51 | 0.57 | 0.52 | 0.56 | 0.49 | 0.61 | 0.55 | 0.54 |
| 2.00 | 0.07 | 0.60 | 0.30 | 0.36 | 0.31 | 0.35 | 0.27 | 0.43 | 0.34 | 0.32 |
| 2.50 | 0.21 | 1.89 | 0.54 | 0.80 | 0.57 | 0.74 | 0.34 | 1.00 | 0.71 | 0.66 |
| 3.00 | 0.70 | 2.25 | 1.10 | 1.25 | 1.12 | 1.22 | 1.00 | 1.40 | 1.21 | 1.17 |
| 3.50 | 0.42 | 1.12 | 0.58 | 0.73 | 0.60 | 0.70 | 0.51 | 0.84 | 0.67 | 0.63 |
| 4.00 | 0.28 | 1.29 | 0.51 | 0.74 | 0.53 | 0.69 | 0.39 | 0.88 | 0.64 | 0.61 |
| 4.50 | 0.20 | 0.88 | 0.36 | 0.50 | 0.37 | 0.49 | 0.28 | 0.63 | 0.45 | 0.42 |
| 5.00 | 0.20 | 0.73 | 0.35 | 0.50 | 0.37 | 0.48 | 0.27 | 0.57 | 0.43 | 0.42 |
| 5.50 | 0.26 | 0.65 | 0.40 | 0.52 | 0.41 | 0.50 | 0.31 | 0.56 | 0.45 | 0.45 |
| 6.00 | 0.31 | 0.72 | 0.49 | 0.59 | 0.51 | 0.59 | 0.39 | 0.63 | 0.53 | 0.54 |
| 6.50 | 0.27 | 0.58 | 0.38 | 0.42 | 0.39 | 0.41 | 0.34 | 0.44 | 0.40 | 0.40 |
| 7.00 | 0.24 | 0.44 | 0.34 | 0.38 | 0.35 | 0.37 | 0.31 | 0.39 | 0.36 | 0.37 |
| 7.50 | 0.24 | 0.40 | 0.33 | 0.36 | 0.33 | 0.35 | 0.30 | 0.37 | 0.34 | 0.35 |
| 8.00 | 0.25 | 0.39 | 0.33 | 0.35 | 0.33 | 0.35 | 0.31 | 0.37 | 0.34 | 0.34 |
| 8.50 | 0.25 | 0.36 | 0.31 | 0.34 | 0.32 | 0.33 | 0.30 | 0.35 | 0.32 | 0.33 |
| 9.00 | 0.24 | 0.36 | 0.31 | 0.33 | 0.31 | 0.33 | 0.29 | 0.34 | 0.32 | 0.32 |
| 9.50 | 0.24 | 0.33 | 0.29 | 0.31 | 0.29 | 0.31 | 0.28 | 0.32 | 0.30 | 0.30 |
| 9.95 | 0.22 | 0.33 | 0.29 | 0.31 | 0.29 | 0.30 | 0.27 | 0.32 | 0.29 | 0.30 |

**Table 15.3.6-1 Aerosol Removal Coefficients as Calculated by Uncertainty Analysis ($hr^{-1}$)**