

15.0 TRANSIENT AND ACCIDENT ANALYSES

15.1 Introduction

In Chapter 15, "Accident Analyses," of the design control document (DCD), Westinghouse, the applicant, described the safety analyses of various design-basis transients and accidents. The results of these analyses demonstrate conformance of the AP1000 design with the acceptance criteria. The acceptance criteria for the design-basis events are based on meeting the relevant requirements and general design criteria (GDC) specified in Appendix A, "General Design Criteria for Nuclear Power Plants," to Title 10 of the Code of Federal Regulations (10 CFR) Part 50, "Domestic Licensing of Production and Utilization Facilities." The specific acceptance criteria for each event appear in NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," issued March 2007 (hereafter referred to as the SRP). The acceptance criteria include GDC 10, "Reactor Design," pertaining to the specified acceptable fuel design limits, and GDC 15, "Reactor Coolant System Design," pertaining to the design conditions of the reactor coolant pressure boundary. These criteria ensure that these limits are not exceeded during any conditions of normal operation, including anticipated operational occurrences. For postulated loss-of-coolant accidents (LOCAs), the applicant analyzed various break sizes and locations to show compliance with the emergency core cooling system (ECCS) performance acceptance criteria specified in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors."

Various subsections and tables in Section 15.0 of the DCD describe the transient and accident events analyzed in Chapter 15 and the computer codes used and assumptions made in the safety analyses of these events. The assumptions relate to the design plant conditions, initial conditions, protection and safety monitoring system (PMS) setpoints and time delays to the reactor trip functions, the actuation of engineered safety features (ESFs), and the plant systems and components available for the mitigation of transients and accidents.

Revision 17 of the DCD includes numerous changes to Section 15.0 of the DCD. The sections below describe the specific changes and their evaluation by the staff of the U.S. Nuclear Regulatory Commission (NRC).

15.1.0.3 Plant Characteristics and Initial Conditions Assumed in the Accident Analyses

15.1.0.3.1 Power Measurement Uncertainty

The initial conditions assumed in the existing Chapter 15 safety analyses include a ± 2 -percent allowance for the power calorimetric error. In Revision 17 of the DCD, Section 15.0.3.2, "Initial Conditions," Table 15.0-2, "Summary of Initial Conditions and Computer Codes Used (Sheets 1-5)" (Footnote a), and Table 15.0-5, "Determination of Maximum Power Range Neutron Flux Channel Trip Setpoint, Based on Nominal Setpoint and Inherent Typical Instrumentation Uncertainties," now include the following sentence:

The main feedwater flow measurement supports a 1 percent power uncertainty; use of a 2 percent power uncertainty is conservative.

The applicant also changed the initial thermal power assumed for the large-break LOCA analysis in Table 15.0-2 from 3,468.0 megawatts thermal (MWt) to 3,434.0 MWt. Based on the AP1000 rated thermal power of 3,400 MWt, the initial thermal power of 3,434 MWt represents a 1-percent allowance for the power calorimetric error.

15.1.0.3.1.1 Evaluation

The review guidance in SRP Section 15.0 states that the reviewer ensures that the application specifies the permitted fluctuations and uncertainties with reactor system parameters and assumes the appropriate conditions, within the operating band, as the initial condition for transient analysis. For analyses of postulated LOCAs, Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 specifies that an assumed power level lower than 1.02 times the licensed power level may be used, provided the proposed alternative value has been demonstrated to account for uncertainties resulting from errors in power-level instrumentation.

The existing Chapter 15 safety analyses for all transients and accidents assume the rated thermal power with a calorimetric error of 2.0 percent as the initial condition. Revision 17 of the DCD continues to assume the 2 percent power uncertainty for all events except for the large-break LOCA; for this event, it assumes the rated thermal power with a 1 percent power uncertainty as the initial condition. The AP1000 DCD does not describe the instrumentation or methodology used for the main feedwater flow measurement, nor does it provide a basis to support the claimed 1 percent power uncertainty.

In its response to Request for Additional Information (RAI)-SRP-15.0-SRSB-02 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML091310260), Westinghouse stated that the AP1000 will use the proven application of high-accuracy ultrasonic feedwater flow measurement and high-accuracy feedwater temperature measurement to affect a high-accuracy plant calorimetric. AP1000 licensees will calculate the plant calorimetric uncertainty and verify that the actual plant instrumentation performance is bounded by the design value of 1 percent calorimetric uncertainty. For traceability, Section 15.0.15, "Combined License Information," will include the following combined license (COL) information item:

15.0.15.1 Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters prior to fuel load, the Combined License holder will calculate the primary power calorimetric uncertainty. The calculations will be completed using an NRC acceptable method and confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated values.

The applicant will also revise DCD Table 1.8-2, "Summary of AP1000 Standard Plant Combined License Information Items," to include COL Item 15.0-1, "Documentation of Plant Calorimetric Uncertainty Methodology," as an action item for the COL holder.

The staff notes that the NRC has approved high-accuracy ultrasonic feedwater measurement instrumentation that can achieve a 1 percent power measurement uncertainty, and this instrumentation has been used as a basis for power uprates for operating plants. The staff concludes that COL Information Item 15.0.15.1 and COL Item 15.0-1 provide acceptable vehicles for the COL license holder to confirm its selected instrumentation for the main feedwater measurement, with the power calorimetric measurement uncertainty bounded by the 1 percent power uncertainty assumed in the large-break LOCA analysis.

The staff identifies the inclusion of COL Information Item 15.0.15.1 in DCD Section 15.0.15 and COL Item 15.0-1 in Table 1.8-2 as Confirmatory Item (CI)-SRP15.0-SRSB-02, to track its incorporation in a future revision of the AP1000 DCD.

15.1.0.3.1.2 Conclusion

Based on the above evaluation, the staff finds that COL Item 15.0-1 and COL Information Item 15.0.15.1 provide acceptable vehicles to confirm the 1-percent power uncertainty assumed in the large-break LOCA analysis. The uncertainty pertaining to the initial condition of the power level is therefore properly addressed, consistent with SRP Section 15.0 and Appendix K to 10 CFR 50.46 regarding LOCAs. Therefore, the staff concludes that the above proposed changes are acceptable, upon successful resolution of CI-SRP15.0-SRSB-02.

15.1.0.3.2 Axial Power Shape

Section 15.0.3.3, "Power Distribution," in Revision 15 of the DCD states that the axial power shape used in the departure from nucleate boiling (DNB) calculation is the 1.55 chopped cosine, as discussed in Section 4.4 for transients analyzed at full power. In Revision 17 of the DCD, the applicant changed "the 1.55 chopped cosine" to "a chopped cosine."

15.1.0.3.2.1 Evaluation

Section 4.4.4.3.2, "Axial Heat Flux Distributions," of the DCD states that the reference axial shape used in establishing core DNB limits (that is, overtemperature ΔT protection system setpoints) is a chopped cosine with a peak-to-average value of 1.61. The staff finds that the applicant made the proposed change in Section 15.0.3.3 of Revision 17 of the DCD to eliminate the specific mention of the 1.55 chopped cosine in order to correct the inconsistency with the value given in Section 4.4.4.3.2 of the DCD. The staff concludes that the proposed change is acceptable because the Chapter 15 safety analyses for the transients were performed with the axial power shape described in DCD Section 4.4.4.3.2.

15.1.0.3.2.2 Conclusion

The staff has reviewed the proposed change in Section 15.0.3.3 of Revision 17 of the DCD to eliminate the specific value of 1.55 for the chopped cosine axial power shape. Based on the above evaluation, the staff concludes that this proposed change is acceptable because the applicant made it in order to correct an inconsistency with the value specified in DCD Section 4.4.4.3.2, which was referenced in DCD Section 15.0.3.3.

15.1.0.6 Protection and Safety Monitoring System Setpoints and Time Delays to Trip Assumed in Accident Analyses

15.1.0.6.1 Change of "High-1" to "High-2" Containment Pressure for "S" Signal or Engineered Safety Feature Actuation

Revision 17 of the DCD changes Table 15.0-4a, "Protection and Safety Monitoring System Setpoints and Time Delay," from having the "S" signal trip function on high-1 containment pressure to its functioning on high-2 containment pressure. In addition, in Table 15.0-6, "Plant Systems and Equipment Available for Transient and Accident Conditions (Sheet 1 of 5)," the applicant changed the ESF actuation function credited for the steam system piping failure,

feedwater system pipe break, and LOCA events from high-1 containment pressure to high-2 containment pressure.

15.1.0.6.1.1 Evaluation

The applicant made the change from “high-1” to “high-2” containment pressure in DCD Tables 15.0-4a and 15.0-6 for the “S” signal trip function and the ESF actuation function credited in the safety analyses, respectively, in order to be consistent with Technical Specification (TS) Table 3.3.2-1, “Engineered Safeguards Actuation System Instrumentation,” which indicates that the safeguard actuation function 1.b is “containment pressure—high 2.” Although the applicant changed the term “high-1 containment pressure” to “high-2 containment pressure,” the value of the setpoint assumed in the safety analyses is not changed, and therefore the safety analyses are not affected. Since this is merely a change in terminology for consistency, the staff concludes that this change is acceptable.

15.1.0.6.1.2 Conclusion

Under 10 CFR 50.36, “Technical Specifications,” the plant TS must specify the limiting safety system settings of automatic protective devices. TS Table 3.3.2-1 for the AP1000 specifies the “Containment Pressure—High 2” trip setpoint for the safeguard actuation function. The staff finds the proposed change from “high-1” to “high-2” containment pressure in DCD Tables 15.0-4a and 15.0-6 to be acceptable because the applicant has made it in order to be consistent with the terminology used in the AP1000 TS, without changing the setpoint value or safety analyses.

15.1.0.6.2 Changes to Pressurizer Water Level Setpoints

In Revision 17 of the DCD, Table 15.0-4a, the applicant changed the limiting setpoint for actuation of passive residual heat removal (PRHR) on the high-3 pressurizer water level from 80 percent of span to 76 percent of span. It also changed the chemical and volume control system (CVS) isolation on the high-2 pressurizer water level from 67 percent of span to 63 percent of span, and the CVS isolation on the high-1 pressurizer water level coincident with “S” signal from 30 percent of span to 28 percent of span.

15.1.0.6.2.1 Evaluation

The applicant changed the high-3, high-2, and high-1 pressurizer water level setpoints because of a design change in pressurizer dimensions. As described in Technical Report (TR) 36, APP-GW-GLR-016, Revision 0, “AP1000 Standard Combined License Technical Report, AP1000 Pressurizer Design,” the applicant has changed the AP1000 pressurizer dimensions to accommodate the space constraint. This dimensional change reduces the height and increases the diameter of the pressurizer but maintains the same overall pressurizer volume. Therefore, in Revision 17 of the DCD, Table 15.0-4a, the setpoints assumed in the safety analyses for the high-3, high-2, and high-1 pressurizer water levels change from 80 percent, 67 percent, and 30 percent to 76 percent, 63 percent, and 28 percent, respectively, to maintain the same water volumes of the corresponding setpoints. The applicant also revised the high pressurizer water level setpoints in TS Tables 3.3.1-1 and 3.3.2-1, respectively, to be consistent with the revised assumed measurement instrumentation uncertainties accounted for in the safety analysis.

In its response to RAI-TR36-012 (ADAMS Accession Number ML063540065), Westinghouse reanalyzed Chapter 15 limiting events using the changed dimension and the revised setpoints. The results show no or minimal effects on the Chapter 15 events, and the applicable

acceptance criteria for each event continue to be met. Therefore, the staff concludes that these changes are acceptable.

15.1.0.6.2.2 Conclusions

Based on the above evaluation, the staff finds that the applicant has made the proposed changes to the high-3, high-2, and high-1 pressurizer water level setpoints in Revision 17 of the DCD, Table 15.0-4a, and in TS Tables 3.3.1-1 and 3.3.2-1 to accommodate the pressurizer dimension change without changing the corresponding water volume. The staff concludes that these changes are acceptable because the applicant's reanalyses of the Chapter 15 limiting events using the revised dimension and setpoints show no more than minimal effects, and the applicable acceptance criteria for each event analyzed continue to be met.

15.1.0.6.3 Change to the Limiting Setpoint of the Boron Dilution Block on the Source Range Flux Doubling Function

In Revision 17 of the DCD, Table 15.0-4a, the applicant changed the limiting setpoint of the boron dilution block on the source range flux multiplication (doubling) function from "1.6 over 50 minutes" to "3.0 over 50 minutes."

15.1.0.6.3.1 Evaluation

The safety analyses of the boron dilution events during shutdown operation credit the boron dilution block on the source range flux multiplication function (DCD Section 15.4.6, "Chemical Volume and Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant"). In Revision 17 of the DCD, Table 15.0-4a, the applicant changed the terminology for the boron dilution block from "source range flux multiplication" to "source range flux doubling." In Section II.9, "Flux Doubling/Boron Dilution Modifications," of TR-80, APP-GW-GLR-080, "Mark-up of AP1000 Design Control Document Chapter 7," Revision 0, Westinghouse discussed the change for the setpoint of the source range flux doubling signal for the boron dilution block. The source range flux doubling signal is used for protection against boron dilution events during shutdown operation. If the source range neutron flux were to increase and exceed the setpoint because of a decrease in boron concentration in the reactor coolant, the boron dilution protection functions would be actuated. Westinghouse stated that an analysis of the source range flux doubling setpoints indicates that there is a significant likelihood that the setpoints could lead to inadvertent actuation of the boron dilution protection actions when the plant is shut down and the flux doubling function is active, even in the absence of actual changes in core neutron multiplication. This occurs because of the variability inherent in counting a discrete random process such as the leakage of neutrons from the reactor core, especially at relatively low count rates. Therefore, Revision 16 of the DCD changed the nominal setpoint from 1.6 over 50 minutes to 2.2 over 50 minutes to reduce the likelihood of inadvertent actuation of the boron dilution protection functions during normal operation.

DCD Section 15.4.6 provides the safety analyses of boron dilution events resulting from CVS malfunction during various modes of operation. The analyses for dilution during cold shutdown (Mode 5), safe shutdown (Mode 4), hot standby (Mode 3), and startup (Mode 2) take credit for the source range flux doubling signal with the increased setpoint that isolates the makeup flow to the reactor coolant system (RCS) from the demineralized water storage tank. The results indicate that automatic protective actions initiate to minimize the approach to criticality and maintain the plant in a subcritical condition.

In Revision 17 of the DCD, TS Table 3.3.2-1, the trip setpoint for function 15.a. also specifies the source range flux doubling setpoint of 2.2 over 50 minutes for ESF actuation system function 15, boron dilution block. The staff questioned the value used in the safety analysis with the consideration of instrumentation uncertainties. In its response to RAI-SRP15.0-SRSB-01 (ADAMS Accession Number ML082140228), Westinghouse stated that the safety analysis cases reported in DCD Section 15.4.6 support a setpoint of 3.0 over 50 minutes to mitigate the boron dilution event in Modes 3, 4, and 5. It further stated that DCD Table 15.0-4a should also report this value of 3.0 as the setpoint assumed in the safety analysis. The applicant has incorporated this change in Revision 17 of the DCD, Table 15.0-4a. In its response to RAI-SRP15.4.6-SRSB-02 (ADAMS Accession Number ML0834000741), Westinghouse stated that the boron dilution analyses performed for Modes 3, 4, and 5 documented in Revision 17 of the DCD, Section 15.4.6, assumed a safety analysis setpoint for the boron dilution protection system of 3.0 over 50 minutes with acceptable results (i.e., the automatic protective actions terminate the boron dilution event and maintain the plant in a subcritical condition). Therefore, the staff concludes that the change in Table 15.0-4a of Revision 17 of the DCD, of the limiting setpoint of the boron dilution block on source range flux doubling to 3.0 over 50 minutes, is consistent with the assumption in the safety analysis and therefore is acceptable.

Regarding the TS nominal setpoint of 2.2 over 50 minutes, Westinghouse stated that it has estimated the major factors that contribute to the measurement uncertainty of the boron dilution algorithm. To ensure that the neutron count integrals are as large as possible, the design features include placing the source range neutron detectors outside the reactor vessel along the cardinal axes of the reactor core with the highest neutron leakage, and increasing the counting interval from 60 seconds to 120 seconds in the algorithm. Westinghouse presumes that the 36 percent difference between the value assumed in the accident analysis and the setpoint selected will bound the value determined when final plant design inputs are available. The staff notes that in DCD Tier 1, Section 2.5.2, "Protection and Safety Monitoring System," Table 2.5.2-8, "Inspections, Tests, Analyses and Acceptance Criteria," design commitment item 10 specifies that the PMS setpoints are determined using a methodology that accounts for loop inaccuracies, response testing, and maintenance or replacement instrumentation. If the measurement uncertainty after the final instrumentation installation is not bounded by the 36-percent allowance, the nominal setpoint would be revised accordingly. Therefore, the staff concludes that there is adequate assurance that the nominal setpoint specified in the TS is consistent with the safety analysis setpoint and the measurement uncertainty.

15.1.0.6.3.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed change of the source range flux doubling setpoint from "1.6 over 50 minutes" to "3.0 over 50 minutes," in Revision 17 of the DCD, Table 15.0-4a, is acceptable because this change is consistent with the assumption made in the safety analysis of the boron dilution events with acceptable results.

15.1.0.6.4 Deletion of P-8 Interlock and Replacement with P-10

In Revision 17 of the DCD, the applicant deleted the "High neutron flux, P-8" reactor trip interlock from Table 15.0-4a, and in Table 15.0-6 it changed the permissive interlock for the power range high flux, low-flow trip function credited for the "startup of an inactive reactor coolant pump at incorrect temperature" event from P-8 to P-10.

15.1.0.6.4.1 Evaluation

The power range nuclear power P-8 interlock permits a reactor trip on low flow or reactor coolant pump (RCP) high bearing water temperature in a single loop. In TR-80, APP-GW-GLR-080, Revision 0, Westinghouse provided the rationale for the deletion of the high neutron flux P-8 interlock from the PMS. As stated in TR-80, Section II.7, "Low Reactor Coolant Flow Reactor Trip Logic Modifications," the AP1000 is not licensed for N-1 operation and therefore the design of the PMS is not required to account for this requirement. The applicant also deleted the P-8 interlock from DCD Table 7.2-3, "Reactor Trip Permissives and Interlocks," and TS Table 3.3.1-1, trip function 10, reactor coolant flow—low, and function 11, RCP bearing water temperature—high. Therefore, the deletion of P-8 from DCD Table 15.0.4a is consistent with the reactor trip logic and TS. The staff concludes that the deletion of the P-8 interlock from DCD Tables 15.04a and 7.2-3 and TS Table 3.3.1-1 is acceptable because the P-8 interlock is not needed for the AP1000 design.

In Revision 15 of the DCD, one of the reactor trip functions used for the "startup of an inactive reactor coolant pump at an incorrect temperature" event is the low-flow trip with P-8 interlock. In Revision 17, the applicant changed the P-8 interlock credited for this event to the P-10 interlock. As discussed above, the applicant deleted the P-8 interlock because the AP1000 is not licensed for N-1 loop operation. The power range nuclear power P-10 interlock, which has a lower setpoint than the P-8 interlock (10 percent and 48 percent power for P-10 and P-8, respectively), performs the same function of allowing reactor trip on low coolant flow or RCP high bearing water temperature in multiple loops that the P-8 interlock performs for single loop. Since the P-10 interlock has lower setpoint than the P-8 interlock, the replacement of the P-8 with the P-10 interlock is conservative because it would permit the reactor to be tripped at a lower power for this event. The staff also requested that Westinghouse confirm that safety analyses of transient events that take credit for the P-8 interlock have been reanalyzed with the P-10 interlock. The staff agrees with the applicant's response to RAI-SRP15.3.1-SRSB-01 (ADAMS Accession Number ML082770191) that the replacement of the P-8 with the P-10 interlock will improve the results of the non-LOCA safety analysis, including for loss-of-flow events, at lower power since the reactor will now be tripped by a partial or completed loss of flow down to 10 percent power.

In Revision 17 of the DCD, TS Table 3.3.1-1, the applicant replaced the P-8 interlock with the P-10 interlock for trip function 10, reactor coolant flow—low, and function 11, RCP bearing water temperature—high. The staff concludes that the change to DCD Table 15.0-6 to replace the P-8 interlock with the P-10 interlock is acceptable because it is consistent with the reactor trip logic and the TS.

15.1.0.6.4.2 Conclusion

The staff has reviewed Revision 17 of the DCD for the removal of the power range nuclear power reactor trip P-8 interlock and replacement with the P-10 interlock where applicable. The staff finds these changes conservative and acceptable because the P-10 interlock has a lower setpoint than the P-8 interlock, and the safety analyses of non-LOCA events credited with this interlock would continue to show compliance with GDC 10 and 15 without exceeding the specified acceptable fuel design limits and the RCS pressure boundary design limit.

15.1.0.6.5 Changes in Valve Opening Time Delay

In Revision 17 of the DCD, Table 15.0-4a, the applicant changed the time delay for the automatic depressurization system (ADS) Stage 1 actuation on core makeup tank (CMT) low level signal from 20 seconds to 30 seconds for the control valve to begin to open. It also changed the ADS Stage 4 actuation on CMT low-low level signal from 30 seconds to 2 seconds for the squib valve to begin to open. In Table 15.0-4b, the applicant changed the closure time of the CVS makeup isolation valves from 10 to 30 seconds, and in Table 15.6.5-10, it also changed the control valve actuation and opening times for ADS Stages 1, 2, 3, and 4. It added a note to these tables, stating the following:

The valve stroke times reflect the design basis of the AP1000. The applicable DCD Chapter 15 accidents were evaluated for the design basis stroke times. The results of this evaluation have shown that there is a small impact on the analysis and the conclusions remains valid. The output provided for the analyses is representative of the transient phenomenon.

15.1.0.6.5.1 Evaluation

In RAI-SRP15.0-SRSB-04, the staff asked Westinghouse to clarify the relationship among the 2-second delay time of the ADS Stage 4 squib valves specified in Table 15.0-4a and the opening time delay sequences for Substage A and Substage B of ADS Stage 4 described in DCD Section 7.3.1.2.4, "Automatic Depressurization System Actuation," and the ADS Stage 4 actuation time delay assumed in the safety analyses. In its response to RAI-SRP15.0-SRSB-04 (ADAMS Accession Number ML091870128), Westinghouse provided the following explanation. In the small-break LOCA analysis in DCD Section 15.6.5.4B, ADS Stage 4 was actuated from the coincidence of CMT level less than the low-2 setpoint, low RCS pressure, and a preset time delay following an open signal initiation for the ADS Stage 3 depressurization valves. The 2-second time delay for ADS Stage 4 actuation on CMT low-2 level identified in Table 15.0-4a represents the accident analysis delay time assumed specifically for the squib valve to begin to open. The accident analysis assumes the 2-second signal processing delay after the CMT low-2 level is reached or the preset time delay after the ADS Stage 3 depressurization valve open signal is generated, whichever is later. This explanation clarifies that the ADS-4 opening time assumed in the safety analysis is for the ADS-4 Substage A depressurization valve opening time, and the Substage B depressurization valve will open after the Stage A valve with a delay time specified in DCD Table 15.6.5-10.

Westinghouse also identified in its response to RAI-SRP15.0-SRSB-04 the following inconsistencies in Revision 17 of the DCD, Chapters 7 and 15, which need to be revised:

- Section 7.3.1.2.4 states that ADS Stage 2 actuation is interlocked with ADS Stage 1, and ADS Stage 3 actuation is interlocked with ADS Stage 2. However, no such interlocks exist, and therefore Section 7.3.1.2.4 requires revision.
- Table 15.0-4a indicates an ADS Stage 1 actuation on CMT low-level signal delay time of 30 seconds. This should be revised to 32 seconds, consistent with DCD Table 15.6.5-10, as it includes a 30-second programmable delay time with a 2-second signal proceeding delay.

- DCD Table 15.0.4b incorrectly lists Table 15.6.5-13 for the ADS valve opening times. The ADS valve opening times actually appear in Table 15.6.5-10.
- Table 15.6.5-10 should be modified to add two notes: (1) a description of the interlock of CMT low-2 level as well as 128 seconds after the ADS Stage 3 actuation signal is generated for initiation of ADS-4 Substage A valves, and (2) that the valve opening time of ADS Stage 4 valves includes an “arm-fire” processing delay.

Westinghouse will correct these inconsistencies in the next revision to the AP1000 DCD. The staff identifies this as CI-SRP15.0-SRSB-04.

Because the PMS and equipment actuation delay times assumed in the safety analyses for various design-basis events may differ from the valve stroke time design-basis values listed in DCD Tables 15.0-4a, 15.0-4b, and 15.6.5-10, the staff asked Westinghouse to list the PMS and equipment with the delay times assumed in the Chapter 15 safety analyses that differ from the design-basis values specified in these tables and to evaluate the impact of the inconsistencies in delay time assumptions on the affected events. In its response to RAI-SRP15.0-SRSB-05 (ADAMS Accession Number ML091870125), Westinghouse listed the PMS actuation functions and valves with different delay times. The list includes the ADS Stage 1 and Stage 4 actuations on CMT low level signals; the ADS Stage 1, 2, 3, and 4 control (depressurization) valves; and the CVS makeup isolation valves. The events analyzed with different delay times include boron dilution, inadvertent CVS actuation, loss of normal feedwater events analyzed with different CVS makeup isolation valve delay times, and the small-break LOCA and inadvertent operation of the ADS with different ADS valve opening time delays. Westinghouse evaluated the effects of the inconsistencies on the safety analyses of the affected events.

The design-basis stroke time for the CVS makeup isolation valves is 30 seconds, but the safety analyses for the inadvertent CVS actuation (DCD Section 15.5.2) and loss of normal feedwater (DCD Section 15.2.7) events assume a 12-second valve closure (which includes a 2-second microprocessor delay and 10-second closure time). Based on the CVS makeup flow rate assumed in the analysis, an additional 20 seconds in the makeup valve isolation time would increase the makeup flow into the RCS by about 15 cubic feet. For these two events, there is sufficient available margin between the pressurizer volume and the maximum pressurizer water volume (see DCD Figures 15.5.2-5 and 15.2.7-6, respectively) to accommodate this water volume without changing the conclusion that the minimum DNB ratio remains above the design-limit value and the RCS pressure remains below 110 percent of the design value. For the boron dilution events, Westinghouse’s response to RAI-SRP9.3.6-SRSB-01 (ADAMS Accession Number ML090430155) provides an evaluation. The longer makeup valve closure time results in no adverse effect for the boron dilution events occurring during Mode 1 and 2 operations because the purge volume of the CVS is not sufficient to return the reactor to criticality. For the events occurring during Mode 3, 4, and 5 operations, the safety analyses described in DCD Sections 15.4.6.2.2 through 15.4.6.2.4 assume a makeup valve closure time of 28 seconds, and there is sufficient margin to accommodate the 2-second delay in the valve closure time.

The only non-LOCA event affected by the ADS is an inadvertent operation of the two ADS Stage 1 trains event, since all other non-LOCA events do not model the ADS valves. The existing analysis described DCD Section 15.6.1 for the inadvertent operation of the ADS Stage 1 valve event assumed a 25-second stroke time, compared to the design-basis value of 40 seconds. As shown in DCD Figure 15.6.1-7, the minimum DNB ratio occurs around

22 seconds, which is before the ADS valves are fully open. Therefore, the longer valve stroke time has no effect on the analysis result and the existing analysis remains valid.

Since the primary role of the ADS is to mitigate a small-break LOCA, the staff asked Westinghouse to evaluate the effects of changes in the ADS valve stroke times on the small-break LOCA analysis. In response to RAI-SRP5.4.6-SRSB-01 (ADAMS Accession Number ML090430156), Westinghouse evaluated the effects of the changes in the ADS Stage 1, 2, and 3 stroke times, as well as the ADS Stage 4 time delay change from 30 seconds to 2 seconds, on the small-break LOCA analysis described in DCD Section 15.6.5.4B, "Small-break LOCA Analyses." The evaluation includes inadvertent ADS actuation, a 2-inch cold leg break, and a direct vessel injection (DVI) break. The results show minimal effects on the safety analysis and that the core remains covered in all cases; thus, the design continues to comply with the acceptance criteria for the ECCS specified in 10 CFR 50.46. Therefore, the staff concludes that the changes in the ADS valve actuation and stroke times specified in DCD Tables 15.0-4a and 15.6.5-10 are acceptable.

It should be noted that in Revision 17 of the DCD, Tier 1, Section 2.1.2, "Reactor Coolant System," and Section 2.3.2, "Chemical and Volume Control System," Westinghouse proposed to change the acceptance criteria for the opening times after receipt of a signal from the PMS of the remotely operated ADS Stage 1, 2, and 3 valves (RCS-V001A/B, RCS-V002A/B, and RCS-V003A/B) and the CVS makeup isolation valves (V090 and V091) specified in Tables 2.1.2-4 and 2.3.2-4, "Inspections, Tests, Analyses, and Acceptance Criteria," for the RCS and CVS, respectively. The staff finds that the revised values are acceptable because they are consistent with the values described in DCD Tier 2, Tables 15.0-4b and 15.6.5-10.

15.1.0.6.5.2 Conclusion

Based on the above evaluation, and upon successful resolution of CI-SRP15.0-SRSB-04, the staff concludes that the proposed changes in Revision 17 of the DCD, Tables 15.0.4a, 15.0.4b, and 15.6.5-10, regarding the delay times of the ADS valves and CVS makeup isolation valves are acceptable because they produce insignificant changes in the results of the analyses for the affected design-basis events, and the acceptance criteria in GDC 10, GDC 15, and 10 CFR 50.46 continue to be met.

15.1.0.6.6 Changes Pertaining to Steam Generator Tube Rupture Analysis

In Revision 17 of the DCD, the applicant made the following changes to Table 15.0-4a:

- Change the high-2 steam generator limiting setpoint from 100 percent to 95 percent of the narrow range level span.
- Add an entry for CMT actuation on pressurizer low-2 water level with a time delay of 2.0 seconds.

15.1.0.6.6.1 Evaluation

In its response to RAI-SRP15.0-SRSB-06 (ADAMS Accession Number ML091970101), Westinghouse confirmed that it made these changes to clarify the DCD documentation in areas identified during the Westinghouse DCD Revision 17 review process. The changes do not represent new assumptions or results and are consistent with the assumptions credited in the

existing Chapter 15 safety analyses. The change of the high-2 steam generator limiting setpoint from 100 percent to 95 percent of the narrow range level span is consistent with the steam generator tube rupture analysis provided in DCD Section 15.6.3. Other events used 100 percent of the narrow range level span, but the steam generator tube rupture analysis provides the limiting setpoint. The addition of the CMT actuation on pressurizer low-2 water level with a time delay of 2 seconds is also consistent with the steam generator tube rupture analysis, which credits this signal. The staff therefore concludes that these changes are acceptable.

15.1.0.6.6.2 Conclusion

Based on the above evaluation, the staff concludes that the proposed changes in Revision 17 of the DCD, Table 15.0-4a, pertaining to the steam generator low-2 setpoint and CMT actuation are acceptable because they are made merely for clarification of the DCD documentation and do not represent new assumptions in the steam generator tube rupture analysis.

15.1.0.8 Plant Systems and Components Available for the Mitigation of Accident Effects

15.1.0.8.1 Change to Chemical and Volume Control System Makeup Suction Isolation for Boron Dilution Mitigation

In DCD Revision 17, Table 15.0-6, "Plant Systems and Equipment Available for Transient and Accident Conditions," the applicant changed the equipment credited for the "chemical and volume control system malfunction which results in a boron dilution" event from "low insertion limit annunciators" to "CVS to RCS isolation valves, makeup pump suction isolation valves, from the demineralized water transfer and storage system."

15.1.0.8.1.1 Evaluation

For the event of boron dilution resulting from CVS malfunction, the AP1000 design takes credit for the source range flux doubling signal (as discussed in Section 15.1.0.6.3 above), which will automatically isolate the unborated water source for boron dilution protection. Revision 17 of the DCD changes the CVS makeup valve realignment for boron dilution protection. The applicant revised DCD Section 9.3.6.4.5.1, "Boron Dilution Events," to state that the CVS is designed to address a boron dilution accident by closing redundant safety-related valves, tripping the makeup pumps, or aligning the suction of the makeup pumps to the boric acid tank, or all three. It revised Section 9.3.6.3.7, "Chemical Volume and Control System Valves," to state that these normally open, motor-operated makeup line containment isolation valves close on a source range flux doubling signal to terminate possible unplanned boron dilution events. The applicant also revised Section 7.3.1.2.14, "Boron Dilution Block," to state the following:

In the event of an excessive increasing rate of source range flux doubling signal, the block of boron dilution is accomplished by closing the chemical and volume control system makeup isolation valves and closing the makeup pump suction valves to the demineralized water storage tanks. This signal also provides a non-safety trip of the makeup pumps. These actions terminate the supply of potentially unborated water to the reactor coolant system as quickly as possible.

Therefore, the staff finds that the change in Table 15.0-6 regarding the use of CVS makeup pump suction isolation valves for the mitigation of boron dilution is consistent with the CVS design. As discussed in Section 15.2.4.6 of this report regarding boron dilution events occurring

during Mode 3, 4, and 5 operations, this design change would terminate the boron dilution event sooner and has no safety significant effect on the consequence of the boron dilution events. Therefore, the staff concludes that this change is acceptable.

15.1.0.8.1.2 Conclusion

Based on the above evaluation, the staff concludes that the change in Revision 17 of the DCD, Table 15.0-6, regarding the change to close the makeup line isolation valves for the termination of boron dilution events is acceptable because it reflects the CVS design change, has no significant effect on the consequence of boron dilution events, and the applicable GDC continue to be met.

15.1.0.8.2 Other Changes for Clarification

In Revision 17 of the DCD, the applicant made the following changes to Tables 15.0-6 and 15.0-8:

- Add the main steam isolation valves, startup feedwater isolation, and accumulators credited for the analyses of the inadvertent opening of a steam generator safety valve and steam system pipe failure events (DCD Sections 15.1.4 and 15.1.5).
- Add the steam generator safety valves for the analysis of the inadvertent operation of the CMT during power operation (DCD Section 15.5.1).
- Add the low steamline pressure ESF actuation functions for the analysis of the CVS malfunction that increases reactor coolant inventory (DCD Section 15.5.2).
- Add the low pressurizer level ESF actuation function for the analysis of the steam generator tube rupture (DCD Section 15.6.3).
- Add an entry to specify that the sample line isolation valves are credited for the failure of small lines carrying primary coolant outside containment (DCD Section 15.6.2).
- Revise the footnote to Table 15.0-8 pertaining to the main steam isolation valve backup valves from stating “moisture separator reheat steam supply control valve” to “moisture separator reheater 2nd stage steam isolation valves.”

15.1.0.8.2.1 Evaluation

In its response to RAI-SRP15.0-SRSB-06 (ADAMS Accession Number ML091970101), Westinghouse confirmed that it made these changes merely to clarify the DCD documentation in areas identified during the Westinghouse DCD Revision 17 review process. The changes are made to be consistent with the assumptions credited in the corresponding DCD sections in existing Chapter 15 safety analyses, and they do not represent new assumptions or results.

15.1.0.8.2.2 Conclusion

The staff has reviewed the above changes in Revision 17 of the DCD, Tables 15.0-6 and

15.0-8. The staff concludes that these changes are acceptable because they are made merely for clarification of the DCD documentation and do not represent new assumptions or affect the results in the existing Chapter 15 safety analyses.

15.1.0.12 Component Failures

In Revision 17 of the DCD, the applicant proposed to delete the following paragraph in Section 15.0.12.1, pertaining to operator action error:

A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure. The error is limited to manipulation of safety related equipment and does not include thought process errors that could potentially lead to common cause or multiple errors.

15.1.0.12.1 Evaluation

In Revision 2 of response to RAI-SRP15.0-SRSB-03 dated July 8, 2010 (ADAMS Accession Number ML101930128), Westinghouse stated that while the Chapter 15 safety analyses do meet the intent of the statement in DCD section 15.0.12.1, the statement was originally removed to prevent confusion about the operator action assumptions made for in the DCD safety analyses. There were no changes to the safety analysis assumptions as a result of the removal of this statement. Based on a review of the statement in DCD section 15.0.12.1, it is determined that the deleted statement will be returned to the DCD, with the punctuation of the statement updated as follows:

“A single incorrect or omitted operator action in response to an initiating event is also considered as an active failure; the error is limited to manipulation of safety related equipment and does not include thought process errors that could potentially lead to common cause or multiple errors.”

Therefore, except for the editorial change in the above statement, no change is made to the certified DCD Revision 15 pertaining to operator actions. The staff finds it acceptable.

15.1.0.12.2 Conclusion

As discussed above, the applicant has determined to retain the statement in the certified DCD Revision 15 pertaining to operator actions, except for an editorial change. This is acceptable. However, it will be identified as CI-SRP15.0-SRSB-03 to track its incorporation into DCD Revision 18.

15.2 Transients and Accident Analysis

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

15.2.2.6 Loss of Alternating Current Power to the Plant Auxiliaries (DCD Tier 2, Section 15.2.6)

In Revision 17 of the DCD, Section 15.2.6.2.1, “Method of Analysis,” the applicant added the statement that “the main feedwater flow measurement supports a 1-percent power uncertainty; use of a 2-percent power uncertainty is conservative.” In addition, it deleted the statement from the assumption used in the analysis that “conservative PRHR heat exchanger heat transfer

coefficients (low) associated with the low flow rate caused by the reactor coolant pump trip are assumed.”

15.2.2.6.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the SRP. These acceptance criteria include GDC 10 and 15, which require that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

DCD Section 15.2.6, “Loss of Power to the Plant Auxiliaries,” describes the analysis of the loss of power to the plant auxiliaries caused by a complete loss of offsite grid, accompanied by a turbine-generator trip. Section 15.2.2.6 of NUREG-1793 described the staff evaluation of this event for compliance with the relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

As stated in Section 15.1.0.3.1 of this report, the confirmation of 1-percent power uncertainty will be performed through COL Item 15.0-1. However, the changes have no effect on the analysis of the loss of alternating current power to the plant auxiliaries because the analysis assumes 2-percent power uncertainty. In addition, the applicant deleted the conservative PRHR heat exchanger heat transfer coefficient to provide more accurate information consistent with the existing analysis of the event. The staff concludes that these changes have no effect on the existing analysis of the event, which continues to show compliance with the relevant acceptance criteria; therefore, the changes are acceptable.

15.2.2.6.2 Conclusion

Based on the above evaluation, the staff concludes that the revisions proposed by Westinghouse to Revision 17 of the DCD, Section 15.2.6.2.1, do not represent new assumptions and have no effects on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

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

Revision 17 of the DCD added the following statement to Section 15.2.7.2.1, “Method of Analysis,” regarding the assumptions used in the analysis: “The main feedwater flow measurement supports a 1 percent power uncertainty; use of a 2-percent power uncertainty is conservative.” The applicant also made editorial changes to Table 15.2-1 to correct the time sequence of the loss of normal feedwater flow event; these changes have no effect on the analysis.

15.2.2.7.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the SRP. These acceptance criteria include GDC 10 and 15, which require that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

DCD Section 15.2.7, "Loss of Normal Feedwater Flow," describes the analysis of the loss of normal feedwater flow. Section 15.2.2.7 of NUREG-1793 described the staff evaluation of this event for compliance with the relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

As stated in Section 15.1.0.3.1 of this report, the confirmation of 1-percent power uncertainty will be performed through COL Item 15.0-1. However, the change has no effect on the analysis of the loss of normal feedwater flow because the analysis assumes 2 percent power uncertainty. Therefore, the staff concludes that this change is acceptable.

15.2.2.7.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by Westinghouse to Revision 17 of the DCD, Section 15.2.7, do not represent new assumptions and have no effects on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

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

Revision 17 of the DCD changes Section 15.2.8.1, "Identification of Causes and Accident Description," to include the high-3 pressurizer water level as a condition for a reactor trip. The applicant also added the statement, "Method of Analysis," that "the main feedwater flow measurement supports a 1 percent power uncertainty; use of a 2 percent power uncertainty is conservative," to Section 15.2.8.2.1.

15.2.2.8.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the SRP. These acceptance criteria include GDC 10 and 15, which require that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

DCD Section 15.2.8, "Feedwater System Pipe Break," describes the analysis of feedwater system pipe break. Section 15.2.2.8 of NUREG-1793 described the staff evaluation of this event for compliance with the relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

In its response to RAI-SRP15.0-SRSB-06 (ADAMS Accession Number ML091970101), Westinghouse stated that it added the high-3 pressurizer water level as a reactor trip actuation

condition in order to provide more accurate information in the DCD to be consistent with the existing safety analysis assumption. The staff finds the change acceptable since it does not represent a new assumption and has no effect on the analysis result.

As stated in Section 15.1.0.3.1 of this report, the confirmation of 1 percent power uncertainty will be performed through COL Item 15.0-1. However, the change has no effect on the analysis of the feedwater system pipe break because the analysis assumes 2 percent power uncertainty. Therefore, the staff concludes that this change is acceptable.

15.2.2.8.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by Westinghouse to Revision 17 of the DCD, Section 15.2.8, do not represent new assumptions and have no effect on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

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

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

In Revision 17 of the DCD, Section 15.3.1, Westinghouse proposed to change the power range nuclear power reactor trip permissive from the P-8 interlock to the P-10 interlock for protection against the partial loss of forced reactor coolant flow event.

15.2.3.1.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the SRP. These acceptance criteria include GDC 10 and 15, which require that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

DCD Section 15.3.1 describes the safety analysis of the partial loss of forced reactor coolant flow event. Section 15.2.3.1 of NUREG-1793 described the staff evaluation of this event. Therefore, the evaluation in this document addresses the staff evaluation of the proposed change described in Revision 17 of the DCD.

DCD Section 15.3.1.1, "Identification of Causes and Accident Description," describes the cause and progression of the partial loss of forced reactor coolant flow event (i.e., trip of one RCP). The low primary coolant flow reactor trip signal provides protection against this event. Revision 15 of the DCD stated that above permissive P8, low flow in either hot leg actuates a reactor trip, and between approximately 10 percent power (permissive P10) and the power level corresponding to P8, low flow in both hot legs actuates a reactor trip. Revision 17 of the DCD revises this to state that above permissive P10, low flow in either hot leg actuates a reactor trip.

As discussed in Section 15.1.0.6.4 of this report, Revision 17 of the DCD deletes the P-8 interlock because the P-8 interlock, which permits reactor trip on low flow or RCP high bearing temperature in a single loop, is not needed for the AP1000 since it is not licensed for N-1 loop operation. In Revision 17 of the DCD, the applicant deleted the P-8 interlock from Table 7.2-3,

“Reactor Trip Permissives and Interlocks.” Therefore, the P-8 permissive interlock is changed to P-10 interlock for the low-flow trip function. The P-10 interlock also replaces the P-8 interlock in TS Table 3.3.1-1, trip function 10, reactor coolant flow—low, and function 11, RCP bearing water temperature—high. Therefore, the change to replace the P-8 interlock with the P-10 interlock is consistent with the reactor trip logic and TS. As stated in its response to RAI-SRP15.3.1-SRSB-01 (ADAMS Accession Number ML082140228), the change from the P-8 interlock to the P-10 interlock will improve the results of the loss-of-flow events at lower powers since the reactor will now be tripped at the P-10 setpoint of 10-percent power, compared to the P-8 setpoint of 48 percent power. As a result of replacing the P-8 permissive with P-10, the reactor trip with low flow in both hot legs when the power level is between approximately 10 percent (permissive P-10) and that corresponding to permissive P8 (48 percent) is not necessary. The staff concludes that the change to DCD Section 15.3.1.1 is acceptable.

15.2.3.1.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by Westinghouse to Revision 17 of the DCD, Section 15.3.1, to replace the P-8 interlock with the P-10 permissive interlock are conservative relative to the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

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

In Revision 17 of the DCD, Section 15.3.2, Westinghouse proposed to change the power range nuclear power reactor trip permissive from the P-8 interlock to the P-10 interlock for protection against the complete loss of forced reactor coolant flow event.

15.2.3.2.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the SRP. These acceptance criteria include GDC 10 and 15, which require that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

DCD Section 15.3.2, “Complete Loss of Forced Reactor Coolant,” describes the safety analysis of the complete loss of forced reactor coolant flow event. Section 15.2.3.2 of NUREG-1793 described the staff evaluation of this event. Therefore, the evaluation in this document addresses the staff evaluation of the proposed change described in Revision 17 of the DCD.

DCD Section 15.3.2.1, “Identification of Causes and Accident Description,” describes the cause and progression of the complete loss of forced reactor coolant flow event (i.e., trip of all four RCPs). The low primary coolant flow reactor trip signal provides protection against this event. Revision 15 of the DCD stated that above permissive P8, low flow in either hot leg actuates a reactor trip, and between approximately 10 percent power (permissive P10) and the power level corresponding to P8, low flow in both hot legs actuates a reactor trip. Revision 17 of the DCD revises this to state that above permissive P10, low flow in either hot leg actuates a reactor trip.

Section 15.2.3.1 of this report discusses the staff evaluation that concludes that the replacement of the P8 permissive interlock with the P10 interlock for the low reactor coolant flow is

acceptable. This same conclusion applies to the complete loss of forced reactor coolant flow. Therefore, the staff concludes that the change to DCD Section 15.3.2.1 is acceptable.

15.2.3.2.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by Westinghouse to Revision 17 of the DCD, Section 15.3.2, to replace the P-8 interlock with the P-10 permissive interlock are conservative relative to the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and 15. Therefore, the staff concludes that the proposed DCD changes are acceptable.

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

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)

In Revision 17 to the DCD, Section 15.4.1.1, Westinghouse proposed to change the high nuclear flux rate reactor trip. This trip function is actuated when the positive rate of change of neutron flux on two out of four nuclear power range channels indicates a rate above a preset setpoint. Previously, this trip function could be manually bypassed after the coincident two out of four nuclear power range channels were manually reset. Westinghouse proposed to no longer allow the manual bypass of the trip function after the manual reset of the coincident two out of four nuclear power range channels.

15.2.4.1.1 Evaluation

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed such that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of anticipated operational occurrences. Control rod withdrawal is an anticipated operational occurrence. The fuel cladding is the first barrier of protection against radioactive release. Meeting GDC 10 ensures that the fuel cladding integrity is not challenged during this anticipated operational occurrence.

GDC 13, "Instrumentation and Control," requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to ensure adequate safety, and the provision of controls that can maintain these variables and systems within prescribed operating ranges. Meeting GDC 13 ensures that the appropriate controls are provided to maintain these variables and systems within the prescribed operating ranges.

GDC 17, "Electric Power Systems," requires that an onsite electric power system and an offsite electric power system be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to ensure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences. Meeting GDC 17 ensures that the fuel cladding integrity is not challenged during an uncontrolled control rod assembly withdrawal in conjunction with a loss of onsite or offsite power.

GDC 20, "Protection System Functions," requires that the protective system automatically initiate the operation of the reactivity control system to ensure that fuel design limits are not exceeded as a result of anticipated operational occurrences. The withdrawal of a control

assembly significantly impacts local fuel pin power and could lead to cladding failure. Measures are required to ensure that an abnormal rod withdrawal is detected and automatically terminated before fuel design safety limits are violated. Meeting GDC 20 ensures that cladding integrity is not challenged during this anticipated operational occurrence.

GDC 25, "Protection System Requirements for Reactivity Control Malfunctions," requires that the reactor protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods. A failure of the reactivity control system that would create an unmitigated withdrawal of a control assembly could lead to cladding failure. Meeting GDC 25 ensures that a power transient fostered from a reactivity addition as a result of a single failure of the reactivity control system will be detected and terminated before challenging the fuel cladding integrity.

The staff has reviewed the proposed change by Westinghouse to no longer allow the manual bypass of the high nuclear flux rate reactor trip function after the manual reset of the coincident two out of four nuclear power range channels. Since the proposed change would no longer allow manual bypassing of the trip function, the proposed change has no effect on the analysis results for the control rod withdrawal from subcritical transient. Therefore, the staff concludes that this proposed change is acceptable.

15.2.4.1.2 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 design change is acceptable because it continues to meet the requirements of GDC 10, 13, 17, 20, and 25.

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

In Revision 17 to the DCD, Section 15.4.3.1, Westinghouse proposed to delete a sentence pertaining to a specific operator action upon the inoperability of the rod deviation alarm to be consistent with the TS. The resolution of the rod position indicator channel is ± 5 percent of span (± 7.5 inches). A deviation of any rod cluster control assembly (RCCA) from its group by twice this distance (10 percent of span or 15 inches) does not cause power distributions greater than the design limits. The deviation alarm alerts the operator to rod deviation with respect to the group position in excess of 5 percent of span. Westinghouse proposed to delete the sentence from the application that states, "If the rod deviation alarm is not operable, the operator takes action as required by the Technical Specification." However, the following paragraph in the application states that if one or more of the rod position indicator channels is out of service, operating instructions are followed to verify the alignment of the nonindicated RCCAs and the operator also takes action as required by the TS.

15.2.4.3.1 Evaluation

GDC 10 requires that the reactor core and associated coolant, control, and protection systems be designed such that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of anticipated operational occurrences. Control rod withdrawal is an anticipated operational occurrence. The fuel cladding is the first barrier of protection against radioactive release. Meeting GDC 10 ensures that the fuel cladding integrity is not challenged during this anticipated operational occurrence.

GDC 13 requires the provision of instrumentation that is capable of monitoring variables and systems over their anticipated ranges to ensure adequate safety, and the provision of controls that can maintain these variables and systems within prescribed operating ranges. Meeting GDC 13 ensures that the appropriate controls are provided to maintain these variables and systems within the prescribed operating ranges.

GDC 20 requires that the protective system automatically initiate the operation of the reactivity control system to ensure that fuel design limits are not exceeded as a result of anticipated operational occurrences. The withdrawal of a control assembly significantly impacts local fuel pin power and could lead to cladding failure. Measures are required to ensure that an abnormal rod withdrawal is detected and automatically terminated before fuel design safety limits are violated. Meeting GDC 20 ensures that cladding integrity is not challenged during this anticipated operational occurrence.

GDC 25 requires that the reactor protection system be designed to ensure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system, such as accidental withdrawal of control rods. A failure of the reactivity control system that would create an unmitigated withdrawal of a control assembly could lead to cladding failure. Meeting GDC 25 ensures that a power transient fostered from a reactivity addition as a result of a single failure of the reactivity control system will be detected and terminated before challenging the fuel cladding integrity.

In its response to RAI-SRP15.4.3-SRSB-01 (ADAMS Accession Number ML082770131), Westinghouse stated that it revised the sentence, "If the rod deviation alarm is not operable, the operator takes action as required by the Technical Specifications," in Revision 16 of the DCD, Section 15.4.3.1, to be consistent with TS 3.1.4 and 3.1.7. Westinghouse also stated that neither of these TS or any other TS require the rod deviation alarm to be operable. Westinghouse also added that Revision 3 of the DCD revised TS 3.1.4 and 3.1.7 to be consistent with NUREG-1431, "Standard Technical Specifications—Westinghouse Plants," Revision 2, issued April 2001. NUREG-1431, Revision 2, removes the rod deviation alarm since it serves as indication only and does not directly relate to the limiting conditions for operation. This is documented in TSTF-110, "Delete SR frequencies based on inoperable alarms," Revision 2 (ADAMS Accession Number ML040490071). AP1000 TS 3.1.4 does require that all shutdown and control rods shall be operable and that individual indicated rod positions shall be within 12 steps of their group step counter demand position. Surveillance Requirement (SR) 3.1.4.1 requires that individual rod positions are verified within alignment limit every 12 hours. Westinghouse stated that performing this verification every 12 hours provides a history that allows the operator to detect that a rod is beginning to deviate from its expected position. In addition, the specified frequency takes into account other rod position information that is continuously available to the operator in the main control room so that during actual rod motion, deviations can immediately be detected. According to Westinghouse, the digital rod position indication system and the bank demand position indication system make rod position information continuously available to the operator in the main control room. TS 3.1.7 requires the digital rod position indication system and the bank demand position indication system to be operable. The digital rod control system maintains a count of steps taken by each rod group and, based on this information, a digital readout of the demanded bank position is provided and the demanded and measured rod position signals are displayed in the main control room. An alarm is generated whenever an individual rod position signal deviates from the other rods in the bank by a preset limit. Westinghouse verifies that the alarm is set with appropriate allowance for instrument error and within sufficiently narrow limits to prevent exceeding core design hot channel factors.

The staff has reviewed the proposed change by Westinghouse to delete the need for operator action if the rod deviation alarm is not operable. Based on its review, the staff finds that this revision does not affect the safety analysis of the RCCA misalignment events described in DCD Section 15.4.3, and the design continues to comply with the relevant requirements. Therefore, the staff concludes that this proposed change is acceptable.

15.2.4.3.2 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 design change is acceptable because it continues to meet the requirements of GDC 10, 13, 20, and 25.

15.2.4.6 Chemical and Volume Control System Malfunctions that Result in a Decrease in the Boron Concentration in the Reactor Coolant (DCD Tier 2, Section 15.4.6)

In Revision 17 of the DCD, Westinghouse proposed to revise Section 15.4.6 in the areas related to the CVS design modifications associated with the mitigation of boron dilution events. These modifications include the alignment of the makeup pump suction, and flux doubling boron dilution block and CVS isolation pertaining to boron dilution events occurring during Mode 3, 4, and 5 operations. In addition, the applicant made changes regarding the dilution flow rates, RCS mixing volume, and critical and shutdown boron concentrations for Mode 3, 4, and 5 operations.

15.2.4.6.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the SRP. These acceptance criteria include GDC 10 and 15, which require that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any conditions of normal operation, including anticipated operational occurrences; and GDC 13 and GDC 26, "Reactivity Control System Redundancy and Capability," pertaining to appropriate instrumentation and reactivity control system redundancy and capability.

DCD Section 15.4.6 describes the safety analysis for the boron dilution event. Section 15.2.4.6 of NUREG-1793 described the staff evaluation of this event for compliance with relevant acceptance criteria. Therefore, the evaluation in this document is limited to the effects of the proposed changes on compliance with the relevant acceptance criteria.

In Section 15.4.6.1, "Identification of Causes and Accident Description," Westinghouse proposed to modify the description of boron dilution events and the general mitigation method to be consistent with the CVS design changes for boron dilution mitigation described in DCD Section 9.3.6, "Chemical Volume and Control System." The existing design currently realigns the makeup pump suction from the demineralized water tank to the boric acid tank to terminate the potential boron dilution, and to begin to reborate the RCS to restore shutdown margin. These actions would initially cause the boron dilution to continue because the volume of water in the makeup line path would still be unborated until borated water from the boric acid tank began to reach the RCS. The function was changed to close the makeup line isolation valves (as well as the demineralized water isolation valves) or trip the makeup pumps to terminate the event as soon as possible. Long-term recovery from the event would then be accomplished

using either a different flowpath with a smaller unpurged volume or by using the makeup line after purging most of the unborated water in it. The applicant also proposed several text changes along with the logic changes that are required to implement this modification.

The staff reviewed Westinghouse's proposed change related to the realignment from the demineralized water tank to the boric acid tank as well as isolation of the makeup flow to the RCS for the termination of boron dilution events in Modes 3, 4, and 5. Since the proposed change would terminate the boron dilution event sooner and has no safety-significant effect on the transient, the staff finds this change acceptable. The text changes associated with the realignment are therefore acceptable.

In Revision 17 to the DCD, Section 15.4.6.2, Westinghouse proposed to delete text that states that in the event of an inadvertent boron dilution transient during Mode 3, 4, and 5 operations, the source range nuclear instrumentation detects "an increase of 60 percent of" the neutron flux and replace it with generic text reading "a sufficiently large increase in" the neutron flux. In its response to RAI-SRP-15.4.6-SRSB-01 (ADAMS Accession Number ML082140228), Westinghouse explained that in Revision 15 of the DCD, the use of the phrase "an increase of 60 percent" reflected the 1.6 multiplier (over 50 minutes) setpoint reported in DCD Table 15.0-4a. In Revision 16 of the DCD, and supplemented by TR-80, APP-GW-GLR-080, Westinghouse proposed to change the multiplier from 1.6 to 2.2 (which would now be an increase of greater than 60 percent). Westinghouse stated that it reanalyzed the event and verified that an increase in the nominal setpoint from 1.6 to 2.2 demonstrates acceptable results and would reduce the likelihood of an inadvertent actuation of the boron dilution protection functions during normal operation.

The applicant changed Revision 16 of the DCD, Table 15.0-4a, to reflect the proposed 2.2 multiplier. The applicant stated that it decided to remove the text in DCD Section 15.4.6 referencing a specific flux increase value in order to simplify any potential future revisions to the text in the event of a subsequent change in the safety analysis setpoint. By reporting the actual numerical value in DCD Table 15.0-4a, the document completely defines the magnitude of the flux increase modeled in the safety analysis; therefore, the applicant stated that there is no need to repeat that same information in multiple locations within the text. The staff has reviewed the applicant's justification for replacing the text referencing "an increase of 60 percent" with generic text reading "a sufficiently large increase." Since DCD Table 15.0-4a will provide a specific value to quantify the phrase, "a sufficiently large increase," this change is acceptable.

However, in its responses to both RAI-SRP-15.4.6-SRSB-01 and RAI-SRP15.0-SRSB-01 (ADAMS Accession Number ML082140228), Westinghouse proposed to increase the multiplier again from 2.2 to 3.0. In its response to RAI-SRP15.0-SRSB-01, Westinghouse stated that the safety analysis cases reported in DCD Section 15.4.6 support a setpoint of 3.0 over 50 minutes to mitigate the boron dilution event in Modes 3, 4, and 5. However, DCD Section 15.4.6 and TR-80, APP-GW-GLR-080, only discuss the increase from 1.6 to 2.2 and do not mention an increase to 3.0. Westinghouse submitted Revision 17 of the DCD, Table 15.0-4a, which includes the new proposed flux rate setpoint of 3.0, but the staff asked Westinghouse to provide additional information regarding the 3.0 multiplier. In response to RAI-SRP15.4.6-SRSB-02 (ADAMS Accession Number ML0834000741), Westinghouse confirmed that the boron dilution analyses performed for Modes 3, 4, and 5 documented in Revision 17 of the DCD, Section 15.4.6, assumed a safety analysis setpoint for the boron dilution protection system of 3.0 over 50 minutes, and analysis results for all cases are acceptable. Westinghouse also justified why the 36-percent allowance between the safety analysis and TS nominal setpoints of 3.0 and 2.2 over 50 minutes is sufficient to bound the actual value determined when final plant

design inputs are available. The staff reviewed the applicant's response confirming that the boron dilution analyses documented in Revision 17 of the DCD, Section 15.4.6, assumed a safety analysis setpoint for the boron dilution protection system of 3.0 over 50 minutes, and it finds this change acceptable. The staff also reviewed the applicant's justification that the 36-percent allowance between the safety analysis and TS nominal setpoints of 3.0 and 2.2 is sufficient to bound the actual value determined when the final plant design inputs are available, and it finds this explanation acceptable.

Additionally, in Revision 17 to the DCD, Section 15.4.6.2, Westinghouse proposed changes to the dilution flowrate, RCS water volume, critical and shutdown boron concentrations, and automatic protective actions initiation time for Mode 3, 4, and 5 operations. The applicant proposed these changes in order to be consistent with the TS, DCD Section 9.3.6, and the assumed conditions for the inadvertent boron dilution event. In its response to RAI-SRP15.4.6-SRSB-03 (ADAMS Accession Number ML083360042), Westinghouse provided additional explanation for the changes in the RCS water volumes for different modes of operation. Westinghouse stated that it recalculated the RCS water volumes using the latest geometric data available, taking into consideration the design changes made up to this point. These design change refinements result in a change of less than 5 percent for Mode 3 and of less than 8 percent for Mode 5. The Mode 4 RCS volume is the same volume as was assumed in the Mode 3 boron dilution calculation, and no credit is taken for the upper head volume in the assumed active mixing volume in the Mode 3, 4, and 5 calculations. The staff reviewed the applicant's rationale for the changes in RCS water volumes for different modes of operation and concludes that these changes are acceptable.

15.2.4.6.2 Conclusion

The staff reviewed the proposed changes described in Revision 17 of the DCD, Section 15.4.6, regarding the CVS design modifications associated with the mitigation of boron dilution events. Based on the above evaluation, the staff concludes that the AP1000 design changes are acceptable because the analyses of boron dilution events continue to meet the requirements of GDC 10, 13, 15, and 26.

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

DCD Section 15.4.8, "Spectrum of Rod Cluster Control Assembly Ejection Accidents," describes the RCCA ejection events resulting from mechanical failure of control rod mechanism pressure housing. 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 affects the consequences of the events.

Revision 17 of the DCD removes the longitudinal and circumferential failures described in Sections 15.4.8.1.1.5 and 15.4.8.1.1.6, respectively. These changes are supported by TR APP-GW-GLE-016.

15.2.4.8.1 Evaluation

Revision 17 of the DCD removes the failure mechanisms outlined in Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 because of a design change to the upper internals. TR APP-GW-GLE-016 provides a technical description of these changes, but the technical justification for removing the failure mechanisms was unclear.

The staff sent RAI-SRP15.4.8-SRSB-01 to the applicant to ask for more information as to why the failure mechanisms previously covered in DCD Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 are no longer considered. In its response to RAI-SRP15.4.8-01, Westinghouse stated that it removed the failure mechanisms from Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 because Section 3.9.4.1.1 covers RCCA failure mechanisms, and these methods were no longer considered credible. The RAI response further explains that the hypothetical failure of a RCCA housing described in DCD Section 15.4.8 is a rapid positive reactivity insertion independent of the specific failure mechanism; therefore, DCD Sections 15.4.8.1.1.5 and 15.4.8.1.1.6 are not needed.

The NRC staff reviewed the RAI response and agrees that as long as the analysis continues to evaluate a rapid positive reactivity insertion resulting from an ejected rod, the requirements of GDC 28, "Reactivity Limits" (as detailed in Regulatory Guide 1.77, "Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors," issued May 1974, and SRP Sections 15.4.8 and 4.2), are met. Therefore, the staff concludes that the proposed change is acceptable.

15.2.4.8.2 Staff Position Related to the Revision of SRP Section 15.4.8

In Section 15.2.4.8 of NUREG-1793, the staff provided a safety evaluation of the RCCA ejection analysis in accordance with SRP Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," Revision 2, issued July 1981. The AP1000 analysis results of the RCCA ejection events initiated at hot full power and at hot zero power demonstrated that the calculated values of the hot spot radially averaged fuel enthalpy are well within the acceptance criterion of 280 calories per gram specified in SRP Section 15.4.8, Revision 2.

After the AP1000 design certification rulemaking, the NRC revised SRP Section 15.4.8. Revision 3 of SRP Section 15.4.8, issued in March 2007, specifies that the number of failed fuel rods used in the radiological evaluation for the rod ejection events be calculated considering the failure mechanisms addressed in SRP Section 4.2. Appendix B, "Interim Acceptance Criteria and Guidance for the Reactivity Initiated Accidents," to SRP Section 4.2, Revision 3, issued March 2007, specifies that, for the reactivity-initiated accidents such as rod ejection accidents in pressurized-water reactors (PWRs), the total number of fuel rods that must be considered in the radiological assessment is equal to the sum of all of the fuel rods failing either (1) the high cladding temperature failure criteria specified or (2) the pellet-cladding mechanical interaction cladding failure criteria specified.

In 10 CFR 52.47(a)(9), the NRC specifies that for applications for light-water reactor (LWR) nuclear power plants, the technical information in the application shall include an evaluation of the standard plant design against the SRP revision in effect 6 months before the docket date of the application. In addition, 10 CFR 52.63(a)(1) specifies that notwithstanding any provision in 10 CFR 50.109, "Backfitting," while a standard design certification rule is in effect under 10 CFR 52.55, "Duration of Certification," or 10 CFR 52.61, "Duration of Renewal," the Commission may not modify, rescind, or impose new requirements on the certified information, whether on its own motion or in response to a petition from any person, unless the Commission determines in a rulemaking that the change meets one of seven criteria. In

10 CFR 52.79(a)(41), the NRC specifies that for applications for LWR nuclear power plant COLs, the technical information in the final safety analysis report shall include an evaluation of the facility against the SRP revision in effect 6 months before the docket date of the application. In addition, 10 CFR 52.98(c)(1) specifies that if the COL references a certified design, then changes to or departures from information within the scope of the referenced design certification rule are subject to the applicable change processes in that rule. Section VIII, "Processes for Changes and Departures," in Appendix C, "Design Certification Rule for the AP600 Design," and Appendix D, "Design Certification Rule for the AP1000 Design," to 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," specifies the processes for changes and departures from the Tier 1, Tier 2, and Tier 2* information, respectively, in the certified design. For example, Section VIII.B.6 specifies that an applicant or licensee who references this appendix may not depart from Tier 2* information without NRC approval.

Accordingly, the staff determined that the AP600 and AP1000 certified standard designs, which the NRC certified before 2007, are not required to comply with Revision 3 of SRP Sections 15.4.8 and 4.2, issued March 2007. For COL applicants or licensees who reference the AP1000 or AP600 certified designs, the NRC staff will review any change or departure from the certified design that requires prior NRC approval as specified in Section VIII of Appendices C and D to 10 CFR Part 52, respectively.

The staff will evaluate the reactivity initiated accidents such as rod ejection accidents based on the acceptance criteria in effect 6 months before docketing the amendment request, such as the interim acceptance criteria specified in Appendix B to SRP Section 4.2, Revision 3, if a change or departure in fuel design or other aspects is proposed that requires a reevaluation of Final Safety Evaluation Report Chapter 4, "Reactor," or Chapter 15, "Accident Analysis."

15.2.4.8.3 Conclusion

The staff concludes that the RCCA ejection analysis meets the acceptance criteria specified in SRP Section 15.4.8, Revision 2, which were in effect for the AP1000 design certification application. Therefore, the RCCA ejection analysis in the DCD is acceptable and can be used by COL applicants referencing the AP1000 standard design. However, the staff will evaluate the RCCA ejection accident based on the acceptance criteria in effect 6 months before docketing the amendment request, such as the interim acceptance criteria specified in Appendix B to SRP Section 4.2, Revision 3, if a change or departure in fuel design or other aspects is proposed that requires a reevaluation of Final Safety Analysis Report Chapters 4 or 15.

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

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

In Revision 17 of the DCD, Westinghouse proposed to revise Section 15.5.1.2 by deleting the sentence, "No single active failure in any of these systems or equipment adversely affects the consequences of the accident."

15.2.5.1.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the SRP. These acceptance criteria include GDC 10 and 15, which require that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

In Revision 17, Section 15.5.1.2, Westinghouse proposed to delete the following sentence:

No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

As stated in DCD Section 15.5.1.2, regarding the PMS actuations for the mitigation of the event, the PRHR heat exchanger removes the core decay heat. The worst single failure assumed is the failure of one of the two parallel isolation valves in the outlet line of the PRHR heat exchanger to open. Therefore, the single active failure is assumed in the safety analysis. In its response to RAI-SRP15.5.1-SRSB-02 (ADAMS Accession Number ML091970101), Westinghouse clarified that it deleted this sentence to remove contradictory information, and the change has no effect on the existing safety analysis assumptions or methodology. Therefore, the staff concludes that this change is acceptable.

15.2.5.1.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by Westinghouse to Revision 17 of the DCD, Section 15.5.1, to delete a contradicting statement are acceptable because there is no effect on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and 15.

15.2.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory (DCD Tier 2, Section 15.5.2)

In Revision 17 of the DCD, Westinghouse proposed to revise Section 15.5.2, "Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory," by deleting the sentence, "No single active failure in any of these systems or equipment adversely affects the consequences of the accident."

15.2.5.2.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the SRP. These acceptance criteria include GDC 10 and 15, which require that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

In Revision 17, Section 15.5.2.2, Westinghouse proposed to delete the following sentence:

No single active failure in any of these systems or equipment adversely affects the consequences of the accident.

As stated in DCD Section 15.5.2.2, regarding the PMS actuations for the mitigation of the event, the PRHR heat exchanger removes the core decay heat. The worst single failure assumed is the failure of one of the two parallel isolation valves in the outlet line of the PRHR heat exchanger to open. Therefore, the single active failure is assumed in the safety analysis. In response to RAI-SRP15.5.1-SRSB-02 (ADAMS Accession Number ML091970101), Westinghouse clarified that it deleted this sentence to remove contradictory information, and the change has no effect on the existing safety analysis assumptions or methodology. Therefore, the staff concludes that the change is acceptable.

15.2.5.2.2 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by Westinghouse to Revision 17 of the DCD, Section 15.5.2, to delete a contradicting statement to be acceptable because there is no effect on the existing safety analysis, and the design continues to comply with the relevant requirements of GDC 10 and 15.

15.2.6 Decrease in Reactor Coolant System Inventory (DCD Tier 2, Section 15.6, Excluding Section 15.6.5)

15.2.6.1 Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the Automatic Depressurization System

In Revision 17 of the DCD, Westinghouse proposed to revise Section 15.6.1, "Inadvertent Opening of a Pressurizer Safety Valve or Inadvertent Operation of the ADS," to change the motor-operated valve stroke times for the ADS valves for Stages 1–3.

15.2.6.1.1 Evaluation

The acceptance criteria for the design-basis events are based on meeting the relevant requirements and GDC specified in Appendix A to 10 CFR Part 50. The specific acceptance criteria for this event appear in the SRP. These acceptance criteria include GDC 10 and 15, which require that the specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary, respectively, are not exceeded during any conditions of normal operation, including anticipated operational occurrences.

In Revision 17, Section 15.6.1.1, "Identification of Causes and Accident Description," Westinghouse proposed the following changes to the motor-operated valve stroke times for ADS valves during this event:

- Change the ADS Stage 1 design opening time from 25 seconds to 40 seconds.
- Change the ADS Stage 2 and 3 design opening times from 70 seconds to 100 seconds.

Westinghouse also proposed to add the following paragraph to clarify the effects of the proposed times above on the analysis results:

The valve stroke times shown in Chapter 15 tables (input/assumptions) reflect the design basis of the AP1000. The accidents addressed in this section were evaluated for these design basis valve stroke times. The results of this evaluation have shown that there is a small impact on the analysis and the

conclusions remain valid. The output provided in this section for the analyses is representative of the transient phenomenon.

The staff reviewed Revision 17 of the DCD, Section 15.6.1, and concludes that the proposed changes provide additional clarification and do not have a safety significant effect on the existing safety analysis of the event. Therefore, the proposed changes are acceptable.

15.2.6.1.2 Conclusion

Based on the above evaluation, the staff finds that the addition of the above paragraph in Revision 17 of the DCD, Section 15.6.1, only provides clarification and does not have a significant effect on the existing safety analysis. Therefore, the staff concludes that the proposed change is acceptable because the relevant requirements of GDC 10 and 15 continue to be met.

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

15.2.6.5.2 Large Breaks (DCD Tier 2, Section 15.6.5.4A)

In Revision 17 of the DCD, the applicant extensively revised Section 15.6.5.4A, "Large-Break LOCA Analysis Methodology and Results." In Revision 15 of the DCD, the applicant performed the best estimate large-break LOCA (BELOCA) analysis using the WCOBRA/TRAC code to calculate thermal-hydraulic transients in the RCS during a postulated large-break LOCA, and it used the HOTSPOT program to calculate the effects of local models on the calculated peak cladding temperature (PCT). The uncertainties associated with the plant input parameters and states, such as initial fluid conditions in the reactor core system and the ECCS boundary conditions, were treated with a response surface method described in WCAP-12945-P-A, "Code Qualification Document [CQD] for Best Estimate LOCA Analysis," Revision 2, issued 1998. In Revision 17 of the DCD, the applicant continued to use the WCOBRA/TRAC and HOTSPOT computer codes for the BELOCA thermal-hydraulic and hot fuel rod analyses. However, it used the Automated Statistical Treatment of Uncertainty Method (ASTRUM) for the statistical treatment of uncertainties, replacing the existing CQD response surface method. The NRC has reviewed and approved ASTRUM, documented in WCAP-16009-P-A, "Realistic Large Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method," issued 2005. Consistent with the CQD methodology, ASTRUM follows the steps of the Code Scaling, Applicability, and Uncertainty methodology. ASTRUM differs from the CQD methodology primarily in the statistical technique used for uncertainty treatment. ASTRUM uses a nonparametric statistical technique applied directly to a random sample of outputs, for example, the PCT, the maximum local oxidation (MLO), and the core-wide oxidation (CWO). These sample outputs are computed by applying Monte Carlo sampling of the inputs to the WCOBRA/TRAC and HOTSPOT calculations. The uncertainties and biases remain the same as in the CQD methodology.

In support of the revised BELOCA analysis in Revision 17 of the DCD, Section 15.6.5.4A, Westinghouse submitted TR APP-GW-GLE-026, "Application of ASTRUM Methodology for Best-Estimate Large-Break Loss-of-Coolant Accident Analysis," Revision 1, issued 2009, which provides a detailed description of the revised analysis.

The staff used SRP Section 15.6.5, "Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks within the Reactor Coolant Pressure Boundary," and Regulatory

Guide 1.157, "Best-Estimate Calculations of Emergency Core Cooling System Performance," issued May 1989, to guide its evaluation of this revised BELOCA analysis.

15.2.6.5.2.1 AP1000 BELOCA Analysis Code Applicability Evaluation

The BELOCA analysis uses the WCOBRA/TRAC code to calculate the effects of initial conditions, power distributions, and global models, and it uses the HOTSPOT code to calculate the effects of local models. The WCOBRA/TRAC code, described in WCAP-12945-P-A, is Westinghouse's best estimate thermal-hydraulic computer code to evaluate the RCS response to a postulated large-break LOCA. Westinghouse developed the code consistent with the guidance provided in Regulatory Guide 1.157 to calculate thermal-hydraulic conditions in the RCS during blowdown and reflood of a LOCA. The code includes the features needed to satisfy the requirements of 10 CFR 50.46(a)(1)(i). The BELOCA analysis of the AP600 standard design used the WCOBRA/TRAC code. Section 21.6.3 of NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," described the staff review of the application of WCOBRA/TRAC for the AP600 application. Revision 15 of the DCD used the "2000 Formulation" of WCOBRA/TRAC referenced in WCAP-15644-P, "AP1000 Code Applicability Report," Revision 2, issued 2004. This is the version approved in the analysis of the AP600 site safety analysis report, with the subsequent discretionary and nondiscretionary changes that Westinghouse reported to the NRC in 1998, 1999, and 2000, as presented in Appendix A to WCAP-15644-P, Revision 2. Section 21.6.3 of NUREG-1793 described the staff evaluation of WCOBRA/TRAC applicability to the AP1000 BELOCA.

The approved WCOBRA/TRAC code version used in the ASTRUM evaluation model described in WCAP-16009-P-A is WCOBRA/TRAC Mod 7, Revision 6, which is different from the 2000 Formulation of WCOBRA/TRAC used in Revision 15 of the DCD. The WCAP-16009-P-A code version includes the discretionary and nondiscretionary changes reported through the 10 CFR 50.46 reporting process.

The HOTSPOT program is a one-dimensional conduction code that models a portion of a fuel rod at the PCT or burst location and takes into account fuel relocation following a burst during a large-break LOCA. In the axial node where fuel rod burst is predicted to occur, the fuel relocation model in HOTSPOT is used to account for the likelihood that additional fuel-pellet fragments from above that elevation may settle into the burst region. Sections 25.4.2.3 and 25.4.2.4 of WCAP-12945-P-A describe the HOTSPOT model, application, and assessment. HOTSPOT uses a simple physical model that allows the effects of uncertainties to be calculated directly by running the model with parameter values that vary randomly according to specified distributions. The parameter uncertainties for the local models consider the power rates, fuel and cladding properties, metal-water reaction rates, and heat transfer coefficients.

15.2.6.5.2.1.1 WCOBRA/TRAC and HOTSPOT Code Modifications

Revision 17 of the DCD uses WCOBRA/TRAC (M7AR7_AP) and HOTSPOT (6.1) as the current code versions for the AP1000 revised BELOCA analysis. These versions differ from the earlier versions used in Revision 15 of the DCD and WCAP-16009-P-A because of additional code modifications identified through 10 CFR 50.46 reporting. For completeness, Appendix A to APP-GW-GLE-026 identifies a total of 35 changes through 10 CFR 50.46 reporting since 1998.

Appendix B to WCAP-16009-P-A identified 19 of the 35 changes and formed the bases for the acceptance of WCOBRA/TRAC Mod 7A, Revision 6, for BELOCA analyses. Westinghouse

evaluated each of the changes and concluded that each error and its correction, or discretionary change did not have a significant impact on the WCOBRA/TRAC results and did not affect the prior assessments and uncertainties. The NRC evaluated these in its safety evaluation report for WCAP-16009-P-A. The agency concluded that these corrections were reasonable and effectual and found them to be acceptable. The remaining 16 changes were incorporated into WCOBRA/TRAC (M7AR7_AP) and HOTSPOT 6.1, the current code versions used for AP1000 BELOCA analyses.

Of the 16 changes, 10 are discretionary changes involving either (1) enhanced input/output, corrections to output edits, or improvement in the automation of running the code cases, or (2) corrections to errors involving an option, model, code, or a code application that was not used for experiment simulations. Westinghouse evaluated each of these changes and concluded that each discretionary change did not have a significant impact on the WCOBRA/TRAC results and did not affect the prior assessments and uncertainties.

One of the nondiscretionary changes concerned an input error resulting in incomplete solution matrix (see 10 CFR 50.46 letter LTR-NRC-04-17, dated March 25, 2004). Two plant-specific calculations were found to be affected by this error. Westinghouse confirmed that correction of the error did not change the fundamental LOCA transient characteristics (e.g., blowdown cooling and reflood turnaround timing and behaviors). The reference double-ended guillotine break was used to develop the PCT assessments for each plant. The test simulation affected by this error was also corrected, and the transient calculation repeated. It was found that the error correction had no significant effect on the calculation results, and the prior validation conclusions remain valid. The correction is used for the AP1000 BELOCA analyses.

The second nondiscretionary change concerned Inconel 690 material properties capability (see 10 CFR 50.46 letter LTR-NRC-04-17). The material properties were revised in 2000, but the annual 10 CFR 50.46 report did not include the change. This capability was reported to correct that omission. The Inconel 690 material properties are only used for replacement steam generator analyses where the tube material has changed. The analyses directly reflect the effect of the material properties.

The staff identified the following three nondiscretionary changes with the potential to affect any of the prior code assessment and uncertainty results, and that have an effect on the AP1000 BELOCA analyses:

(1) Revised Blowdown Heatup Uncertainty Distribution

This nondiscretionary change was reported in 10 CFR 50.46 letter LTR-NRC-05-20, dated April 11, 2005. This error was previously reported in Westinghouse LTR-NRC-04-11, dated February 3, 2004. As a result of input errors in the loss-of-fluid test (LOFT) facility model used to compare the predicted PCT to the test data to determine this distribution, revised analyses were performed with the version of WCOBRA/TRAC available at that time. As a result of the reanalysis with the modeling error corrections, revised blowdown heatup heat transfer multipliers were developed and the revised cumulative distribution function (CDF) was programmed into the new version of HOTSPOT. Westinghouse estimated the PCT effect of the revised blowdown heatup CDF by calculating the impact on the reference transient for representative two-, three-, and four-loop plants. The estimates bounded all of the 95th percentile HOTSPOT results. Westinghouse also made plant-specific estimates of the effect of the revised overall code uncertainty for blowdown for those plants that track the blowdown period.

Westinghouse identified the errors in the LOFT analyses in its response to RAI-SRP15.6.5-SRSB-03 (ADAMS Accession Number ML083640469). The most important input error was the flag for the fuel rod gap pressure calculation, which was erroneously set to the steady-state option during the transient calculation. With this flag, the critical heat flux calculation was skipped and the transition to film boiling was only a result of the depletion of liquid from the core region. This was the primary input error that impacted the blowdown heatup heat transfer multiplier distribution. Other modeling aspects of the accumulator and break noding were also updated for consistency with the final, approved version of WCAP-12945-P-A, the one-dimensional pipe to three-dimensional vessel connection input was corrected, and errors in the choked flow flag applied to selected components were corrected.

The development of the blowdown heatup heat transfer multipliers and the resulting revised CDF were consistent with the previously approved methods used in WCAP-12945-P-A. The AP1000 ASTRUM analysis uses this revised CDF for the blowdown heatup heat transfer multipliers, consistent with an ASTRUM analysis of a standard Westinghouse PWR.

(2) Improved Automation of End of Blowdown Time

This discretionary change was reported in 10 CFR 50.46 letters LTR-NRC-05-20 and LTR-NRC-06-8, dated March 16, 2006. The automated selection of the end of blowdown time was first modified by replacing the criterion of 40 pounds-force per square inch absolute (psia) with a selection based on the time at which the system pressure stops decreasing. It was again modified by replacing the criterion related to when the system pressure stops decreasing with a selection based on the time when the collapsed liquid level in the lower plenum reaches a minimum and begins to increase again. The redefinition of the end of blowdown is a discretionary change. It has no impact on the WCOBRA/TRAC results and does not affect the prior assessments and uncertainties.

In its response to RAI-SRP15.6.5-SRSB-04 (ADAMS Accession Number ML083640469), Westinghouse confirmed that the AP1000 BELOCA ASTRUM analysis used the improved automated method to define the end of blowdown based on the time at which the collapsed liquid level in the lower plenum reaches a minimum and begins to increase again. The time at which the collapsed liquid level is at its absolute minimum is selected as the end of blowdown time. For large double-ended guillotine breaks, similar results are obtained whether the end of blowdown is defined by RCS pressure criteria or at the time the lower plenum collapsed liquid level reaches a minimum. For the smaller breaks sampled in the ASTRUM analyses, the improved definition based on the lower plenum collapsed liquid level is more applicable than the historical pressure criterion. The ASTRUM analyses use a consistent definition of end of blowdown as the time at which the lower plenum collapsed liquid level is at its absolute minimum; this definition is applied for all runs.

(3) HOTSPOT Fuel Relocation

This nondiscretionary change was reported in 10 CFR 50.46 letter LTR-NRC-08-24, dated May 15, 2008. It was discovered that the effect of the fuel relocation on the local linear heat rate was being calculated correctly in accordance with the approved model,

but then canceled out later in the coding. The HOTSPOT fuel-relocation error was an error in code logic that does not affect the approval of the fuel-relocation model. Westinghouse evaluated the impact of this error in letter DCP/NRC2074, "10 CFR 50.46 Report for the AP1000 Standard Plant Design," dated February 15, 2008. The impact was estimated to be 0 degrees F during the blowdown phase and 70 degrees F during the reflood phase of the accident. The HOTSPOT fuel relocation error is a nondiscretionary change. It has no impact on the WCOBRA/TRAC results and does not affect the prior assessments and uncertainties. The AP1000 BELOCA analyses use the corrected logic.

Westinghouse has incorporated all of the 10 CFR 50.46 discretionary and nondiscretionary changes into the current versions of the WCOBRA/TRAC (M7AR7_AP) and HOTSPOT 6.1 codes. The discretionary changes were shown not to have a significant impact on the WCOBRA/TRAC results and did not affect the prior assessments and uncertainties. The nondiscretionary changes addressed and corrected known errors.

15.2.6.5.2.1.2 WCOBRA/TRAC Code Validation

The phenomena identification ranking table for large-break LOCAs, provided in WCAP-15644-P, Revision 2, and NUREG-1793, indicates that the main difference between the AP1000 and operating PWRs is the DVI. Similar to validation performed for the AP600 design documented in WCAP-14171 and WCAP-14172 (nonproprietary), "WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident," Revision 2, additional validation was performed to examine code capability of WCOBRA/TRAC (M7AR7_AP) to model the AP1000 DVI into the downcomer. Appendix B to APP-GW-GLE-026 documents the DVI assessment calculations. Researchers validated the DVI injection by comparing the code calculations with the tests carried out on the Cylindrical Core Test Facility and the Upper Plenum Test Facility.

Calculations for the Cylindrical Core Test Facility Test 78, Run 58, showed a reasonable prediction of the thermal-hydraulic behavior with WCOBRA/TRAC (M7AR7_AP). The filling of the core and downcomer were in reasonable agreement with the data, and the calculated core level was slightly overpredicted and the downcomer level underpredicted. The maximum clad temperatures in the core were also reasonably predicted at most elevations, and they tended to be overpredicted for the higher power rods. Overall, the Westinghouse evaluation indicated that the WCOBRA/TRAC M7AR7_AP calculations adequately captured the thermal-hydraulic effects associated with downcomer injection, with results similar to those for the AP600.

Calculations for the Upper Plenum Test Facility Test 21, Phases A, B-I, and B-II/III, also showed that the WCOBRA/TRAC (M7AR7_AP) code reasonably and conservatively predicted the downcomer bypass phenomenon, with results similar to those for the AP600.

The Westinghouse assessments demonstrated that the end-of-bypass remains conservatively calculated. They also demonstrated that the PCT predictions from WCOBRA/TRAC (M7AR7_AP) remain conservative at higher fuel rod elevations and in higher power rods, and essentially best estimate at lower elevations.

Based on its review, the staff concludes that WCOBRA/TRAC (M7AR7_AP) conforms to the guidance provided in Regulatory Guide 1.157 and is acceptable for AP1000 BELOCA analyses to demonstrate compliance with the 10 CFR 50.46 criteria.

15.2.6.5.2.2 WCOBRA/TRAC Nodalization Model for AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Analyses

Appendix C to APP-GW-GLE-026 describes the WCOBRA/TRAC nodalization model for AP1000 BELOCA analyses. The nodalization model was developed using the methodology established in WCAP-12945-P-A. The major portion of a BELOCA analysis involves generating the plant-specific vessel and loop model for the WCOBRA/TRAC analyses. The vessel model, in particular, requires detailed information regarding the reactor pressure vessel internals. The AP1000 nodalization was also compared to the available modeling guidance presented in WCAP-16009-P-A.

15.2.6.5.2.2.1 Reactor Coolant System Loop Model

The AP1000 WCOBRA/TRAC RCS loop model (Figure C-3 in APP-GW-GLE-026) uses 75 components and 87 junctions to model the RCS loops and the passive safety systems, including the vessel module and the interface junctions between the one-dimensional loop components and the three-dimensional vessel module. The model includes the four cold legs, the two hot legs, the two steam generators, and the pressurizer.

The two passive safety injection system loops are modeled separately, complete with the CMT and balance line and the accumulator. The model also includes the in-containment refueling water storage tank (IRWST), the sump connections into the DVI lines, the PRHR, and the ADS.

The ADS modeling is simplified and excludes ADS Stages 1, 2, and 3, located on the top of the pressurizer, because ADS Stage 1 is actuated by a low-level signal from the CMTs, whose flow is effectively shut off by the actuation of accumulators. The CMT liquid level is expected to remain above the ADS Stage 1 actuation setpoint throughout the AP1000 BELOCA cladding temperature excursion, even though CMT injection begins again later in the transient. Therefore, ADS Stages 1, 2, and 3 are not expected to actuate during a large break until long after the PCT is calculated to occur. ADS Stage 4, located on the hot legs, is modeled because the limiting WCOBRA/TRAC case is extended beyond the fuel rod quench time until the CMT liquid level decreases to the low-2 setpoint that actuates the ADS Stage 4 valves and IRWST injection.

The RCS loop model also includes one-dimensional PIPE components, which model the thimble tube bypass in the low-power, support column/open hole, and guide tube assemblies. The inclusion of these components is consistent with the approved modeling of the thimble tube bypass in standard Westinghouse PWRs.

WCAP-16009-P-A describes the cold leg break nodalization used to model the break type (guillotine or split) and size.

It should be noted that Revision 17 of the DCD increased the inside diameter of the pressurizer and decreased the vessel height. However, the internal free volume of the pressurizer and the water volume, at power, remain unchanged (refer to APP-GW-GLR-016) and therefore have no effect on the WCOBRA/TRAC loop model. However, the decrease in the pressurizer vessel height results in the changes to the high-3, high-2, and high-1 pressurizer water level setpoints shown in DCD Table 15.0.4a. In addition, in letter DCP/NRC2074, Westinghouse reported that the pressurizer surge-line resistance used in the DCD Revision 15 WCOBRA/TRAC model was in error. This error has been corrected in this WCOBRA/TRAC model.

15.2.6.5.2.2.2 Reactor Pressure Vessel and Internals Model

For the AP1000 17x17 fuel, each fuel bundle contains 264 fuel rods, 24 thimble tubes, and 1 instrumentation tube. The AP1000 reactor pressure vessel model includes five fuel rod groups, based on the upper internal region design and power levels. Rod 1 represents a single fuel rod, the hot rod, which has the highest power in the core and is located in the hot assembly. Rod 2 represents the remaining 263 fuel rods in the hot assembly. It has a power equivalent to the hot assembly average fuel rods and represents an assembly located beneath a support column. Rod 3 represents the 15,576 fuel rods in the medium-power assemblies located beneath support-column/open-hole channels. Rod 4 represents the 18,216 fuel rods in the medium-power assemblies located beneath guide-tube channels. Rod 5 represents the 7,392 fuel rods in the peripheral assemblies in the low-power channel.

The location of the vertical section boundaries relative to the reactor pressure vessel structures and internals were chosen consistent with the approved AP600 WCOBRA/TRAC model described in WCAP-14171, Revision 2; NUREG-1512; and WCAP-14601, "AP600 Accident Analyses—Evaluation Models," Revision 2, with the elevations appropriate for the AP1000 design.

The model includes changes to reflect the following design changes to AP1000 reactor internals described in WCAP-16716-NP, "AP1000 Reactor Internals Design Changes," Revision 2, issued May 2007:

- relocation of radial support keys and tapered periphery on the lower core support plate (LCSP)
- addition of a flow skirt in the lower reactor vessel head
- addition of four neutron panels, attached to the outside surface of the core support barrel

Westinghouse has processes that identify plant configuration changes that could potentially impact safety analyses. It used these internal processes, along with internal processes for assessing evaluation model changes and errors, to identify the need to assess the impacts on LOCA analyses.

As shown in Appendix C to APP-GW-GLE-026, the downcomer region used six azimuthal sectors, as compared to four for the reference model in WCAP-16009-P-A, to account for the DVI lines, similar to the AP600 model. The two additional sectors span two of the four neutron panels, with the remaining two neutron panels falling on gaps (centered between two azimuthal sectors).

The following describes each of the three design changes in detail:

(1) Radial Support Keys/Tapered Periphery on the LCSP

The change to the AP1000 LCSP relocated the four lower radial support keys for the core barrel from the current location of 45 degrees from the cardinal axes to the cardinal locations, which eliminates the potential for interference with the core shroud attachment studs and nuts at the 45-, 135-, 225-, and 315-degree locations. The radial support keys are now physically aligned with the locations of the cold leg nozzles.

In its response to RAI-SRP15.6.5-SRSB-06 (ADAMS Accession Number ML083640469), Westinghouse provided a detailed description of the revised LCSP region modeling for the AP1000 WCOBRA/TRAC model. The metal volume of the LCSP was accounted for in the lower plenum channel. The radial keys were modeled in the downcomer channels to reflect the physical location of the radial keys consistent with the vessel volumes represented by the downcomer channels. One radial key was modeled in each of the two downcomer channel stacks connected to the DVI lines. One half of a radial key was modeled in each of the four downcomer channel stacks connected to the cold legs. The fraction of the radial key in each of these downcomer channels was averaged across the channel because the WCOBRA/TRAC code was not designed to reflect more detailed azimuthal modeling than that specified by the user through the nodalization and lateral gap connections. The momentum area and loss coefficients were adjusted to calibrate the steady-state pressure drop consistent with the modeling approach used for standard plants and the resolution of the WCOBRA/TRAC model nodalization. The AP1000 ASTRUM steady-state calculation confirmed that the steady-state acceptance criteria, specified in WCAP-16009-P-A, Table 12-6, "Criteria for Acceptable Steady-State," were met. The vessel pressure drop and the vessel inlet nozzle to mid-core pressure drop were benchmarked against hydraulic calculations to be within the acceptance criteria.

(2) Lower Reactor Vessel Head Flow Skirt

One of the design changes to the AP1000 reactor internals involved the addition of a flow skirt in the lower reactor vessel head to obtain a more uniform core inlet flow distribution that meets specifications established by the Westinghouse fuel group. Westinghouse provided a detailed description of the flow skirt region modeling for the AP1000 WCOBRA/TRAC model in response to RAI-SRP15.6.5-SRSB-07 (ADAMS Accession Number ML083590305). The metal volume of the flow skirt was accounted for in the lower plenum channel. The radial flow area through the flow skirt holes and between the top of the flow skirt and the bottom of the LCSP was reflected in the gaps that connect the downcomer channels to the lower plenum channel. The AP1000 ASTRUM steady-state calculation confirmed that the steady-state acceptance criteria, specified in WCAP-16009-P-A, Table 12-6, were met. The vessel pressure drop and the vessel inlet nozzle to mid-core pressure drop were benchmarked against hydraulic calculations to be within the acceptance criteria. Westinghouse determined that the core inlet flow distribution assumptions that formed the basis of the original fuel thermal-hydraulic calculations remain valid.

(3) Neutron Panels

Neutron panels were attached to the outside diameter of the core support barrel to maintain the end-of-life reactor vessel fluence values at less than the maximum allowed in Regulatory Guide 1.99, "Radiation Embrittlement of Reactor Vessel Materials," Revision 2, issued May 1988. The panels reduce the reactor vessel fluence at the circumferential locations that have the highest fluence values and provide a relatively rigid structure with a smaller downcomer cross-sectional area than a full cylinder. The neutron panels are located at four circumferential locations where fuel assemblies are closest to the reactor vessel (the 0-, 90-, 180-, and 270-degree locations). The DVI lines are located at 0 and 180 degrees, and the hot leg nozzles are located at 90 and

270 degrees. Each neutron panel covers about 30 degrees circumferentially and extends over the entire length of the active core region (14 feet). The neutron panels are contoured to minimize the impact on the downcomer annulus flow area reduction and to reduce the probability of vortex generation in the downcomer.

Westinghouse provided a detailed description of the neutron panel modeling for the AP1000 WCOBRA/TRAC model in response to RAI-SRP15.6.5-SRSB-09 (ADAMS Accession Number ML083640469). The neutron panels were modeled in the downcomer channels to reflect the physical location of the neutron panels consistent with the vessel volumes represented by the downcomer channels. One neutron panel was modeled in each of the two downcomer channel stacks connected to the DVI lines. One half of a neutron panel was modeled in each of the four downcomer channel stacks connected to the cold legs. The fraction of the neutron panel in each of these downcomer channels was averaged across the channel because the WCOBRA/TRAC code was not designed to reflect more detailed azimuthal modeling than that specified by the user through the nodalization and lateral gap connections. The azimuthal flow areas were modeled as the flow area away from the neutron panels, and the friction factor for azimuthal flow reflects the flow between two walls, consistent with the modeling approach for standard plants. The metal mass of the neutron panels was modeled as an unheated conductor in the appropriate channels. This is an acceptable approach for capturing the effects of the neutron panels in the downcomer, including the calculation of downcomer boiling, and is consistent with the modeling of neutron panels in standard two-loop, three-loop, or four-loop Westinghouse PWRs.

15.2.6.5.2.2.3 Previous WCOBRA/TRAC Modeling Limitation

The AP600 safety evaluation in NUREG-1512, on the acceptability of the AP600 WCOBRA/TRAC model, imposed limitations that require an applicant to address the sensitivity of the CMT and residual heat removal system modeling parameters that are not included in the uncertainty methodology in the event that either the blowdown or reflood phase PCT exceeds 941 degrees C (1725 degrees F) for any reason. The applicant addresses this by repeating the study that identifies the PCT sensitivity to CMT/PRHR elimination and adding the blowdown and reflood PCT impacts as a bias to their respective 95 percent PCT results. Section 15.2.6.5.2 of NUREG-1793 also discussed this limitation. Revision 17 of the DCD, Section 15.6.5.4A.5, states that previous AP1000 sensitivity calculations evaluated the sensitivity of the CMT and PRHR to modeling relative to a baseline case. The results show that the calculated PCTs for both the cases, in which the CMT and the PRHR, respectively, were isolated from the rest of the AP1000, were lower than the PCT of the baseline case. In RAI-SRP15.6.5-SRSB-12, the NRC asked Westinghouse to clarify whether the sensitivity studies for the proposed AP1000 model include the MLO and CWO sensitivities. In its response (ADAMS Accession Number ML083640473), Westinghouse performed sensitivity studies for the CMT and PRHR with the model described in Appendix C to APP-GW-GLE-026. The WCOBRA/TRAC PCT results of the CMT inoperable study showed a temperature decrease of 38 degrees F when compared to the reference case. The results of the PRHR inoperable study showed a temperature increase of 2 degrees F when compared to the reference case. However, to perform the PRHR study, the maximum allowable time step had to be reduced by 0.0001 second to execute the case, and a revised reference case was also run with the reduced time step. The revised reference case resulted in a decreased PCT of 6 degrees F as compared to the original reference case. Westinghouse concluded that the effect of PRHR inoperability was minimal. Since the AP1000 shows significant margin to the MLO and CWO

limits, the staff concludes that no penalties need to be applied to the analysis results for assuming that the safety-related equipment does not operate.

In Attachment 2 to APP-GW-GLE-026, Westinghouse described the following revisions to Revision 17 of the DCD, Section 15.6.5.4A.5, "Large-Break LOCA Analysis Results," to incorporate the results of these studies:

The large break LOCA analysis complies with the restrictions in Reference 32 [WCAP-16009-P-A]. AP1000 sensitivity calculations evaluated the sensitivity to modeling of the CMT and PRHR relative to the reference transient configuration. A case in which the CMT was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was lower than the PCT of the reference transient configuration. Also, a case in which the PRHR was isolated from the rest of the AP1000 was analyzed, and the calculated PCT was 2 °F higher than the reference transient configuration.

This is acceptable; however, the submission of the revised version of Revision 17 of the DCD as stated above is CI-SRP15.6.5-SRSB-01.

In summary, the AP1000 ASTRUM BELOCA model was developed consistent with the modeling guidelines presented in WCAP-16009-P-A. Therefore, the staff concludes that the AP1000 ASTRUM BELOCA model using WCOBRA/TRAC (M7AR7_AP) conforms to the guidance provided in Regulatory Guide 1.157 for evaluation models needed to demonstrate compliance with the acceptance criteria in 10 CFR 50.46.

15.2.6.5.2.3 ASTRUM AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Analysis

15.2.6.5.2.3.1 ASTRUM Applicability to AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Evaluation

ASTRUM uses a nonparametric statistical technique applied directly to a random sample of outputs, for example, the PCT, the MLO, and the CWO. These sample outputs are computed by applying Monte Carlo sampling of the inputs to the WCOBRA/TRAC calculations. With this approach, a simple singular statement of uncertainty in the form of a tolerance interval for the numerical acceptance criteria of 10 CFR 50.46 can be developed. Once the desired tolerance level is defined, the number of Monte Carlo code runs required to construct the tolerance interval that meets the desired level of safety can be computed. ASTRUM is based on a 95/95 tolerance level to demonstrate compliance with the 10 CFR 50.46 criteria. This tolerance level requires 124 WCOBRA/TRAC runs.

A confidence interval covers a population parameter with a stated confidence, that is, a certain proportion of the time. A fixed proportion of the population can also be covered with a stated confidence, called a tolerance interval. The endpoints of a tolerance interval are called tolerance limits. An application of tolerance intervals involves comparing specification limits with tolerance limits that cover a specified proportion of the population. The 95/95 statement is interpreted to mean that there is a 95 percent probability that 95 percent of the random output variable population falls within the specified tolerance limits. The maximum and minimum value of the samples of the output variables are used to define the limits. The results obtained with ASTRUM are taken to mean that there is at least a 95 percent confidence that the limiting PCT, MLO, and CWO from the sample exceed the true 95th percentile.

ASTRUM replaced the original CQD response surface method, described in WCAP-12945-P-A, to combine the uncertainties with a direct Monte Carlo sampling method. In the application of CQD methodology to AP1000, the uncertainties in the initial fluid conditions in the RCS and the ECCS boundary conditions were bounded in a direction to maximize the PCT. The model uncertainty component addressed uncertainties in the code models that affected the overall system transient (global models), as well as those that only affected the hot rod (local models). WCOBRA/TRAC was used to calculate the effects of initial conditions, power distributions, and global models, and HOTSPOT was used to calculate the effects of local models. Biases and uncertainties, resulting from the assumption that the initial conditions, the power distribution, and the model uncertainty components were linearly combined, were quantified and taken into account. The CQD methodology calculates the final PCT uncertainty distribution by a combination of response surface equations and Monte Carlo sampling. ASTRUM considers the same plant parameters, but each parameter is randomly sampled for each case. The 95/95 PCT is established using nonparametric order statistics.

With ASTRUM, the number of runs is fixed (124 runs for three outcomes—PCT, MLO, and CWO) and is independent of the number of uncertainty attributes considered in the sampling process. The uncertainty parameters are directly sampled instead of using the bounding approach of the CQD methodology.

ASTRUM retains the distinction between global and local variables. However, in ASTRUM, only a single HOTSPOT calculation is performed for each WCOBRA/TRAC run, instead of the multiple HOTSPOT runs with the CQD methodology used to obtain the local model PCT distribution. The HOTSPOT calculation is now a single calculation where the local uncertainties are set at their values by random sampling from their respective distributions. This is consistent with the Monte Carlo approach, where each uncertainty parameter is randomly sampled from the respective distribution for each simulation, which comprises a WCOBRA/TRAC and a HOTSPOT calculation.

The NRC reviewed the ASTRUM uncertainty methodology and found it to be acceptable for meeting the regulatory requirements of 10 CFR 50.46, as described in the staff evaluation of WCAP-16009-P-A. The AP1000 BELOCA analyses for Revision 17 of the DCD use the previously approved global model uncertainties and biases, as well as the local model uncertainties and biases, including the revised blowdown heatup transfer multiplier. The ASTRUM uncertainty methodology is independent of the physical system being modeled and is equally applicable to the AP1000 BELOCA analyses to demonstrate compliance with the requirements of 10 CFR 50.46.

15.2.6.5.2.3.2 Application of ASTRUM to the AP1000 Best Estimate Large-Break Loss-of-Coolant Accident Evaluation

The ASTRUM uncertainty methodology, used for the AP1000 BELOCA analysis for Revision 17 of the DCD, independently samples the uncertainties of the global models, local models, power distribution, and initial and boundary conditions for each of 124 runs over the same ranges of uncertainty and distributions as in the CQD methodology. The sampled uncertainties become inputs to each of the 124 WCOBRA/TRAC calculations.

The WCOBRA/TRAC thermal-hydraulic boundary conditions for the hot rod are input to the local model uncertainties calculation performed with HOTSPOT. The limiting PCT, MLO, and CWO may come from the same case or as many as three different cases. With ASTRUM, each

parameter is assumed to be independent of the other two parameters. This assumption is conservative since the MLO and the CWO depend on the cladding temperature (time at temperature).

The WCOBRA/TRAC studies were performed for the AP1000 BELOCA to determine the sensitivities to some of the major plant parameters. These studies included effects of ranging the steam generator tube plugging, ranging the relative power in the low-power assemblies, loss of offsite power coincident with the break initiation, and the break location. The results were used to identify bounding conditions for these parameters, which were then used in the uncertainty calculations.

APP-GW-GLE-026, Table E-1, "Summary of Plant Physical Description, Initial Conditions, Power Distribution, and Global Model Uncertainty Application in CQD Methodology, AP1000 DCD Methodology, and ASTRUM Methodology as Applied to the AP1000," compares the plant physical models, initial conditions, power distributions, and global model uncertainties between the CQD method used in Revision 15 of the DCD and ASTRUM used in Revision 17. The values were referenced to the appropriate information in WCAP-16009-P-A.

DCD Table 15.0-4a provided the PMS setpoints and time delay assumed in the accident analyses. As described in Section 15.2.6.5.2.2 of this report, the safety system related to pressurizer water level high-3, high-2, and high-1 setpoints in DCD Table 15.0-4a are consistent with the revised pressurizer dimensions described in APP-GW-GLR-016.

Revision 17 of the DCD, Table 15.6.5-4, provides the major plant parameter assumptions used in the BELOCA analysis. These plant parameters were developed consistent with the guidance provided in WCAP-16009-P-A. The uncertainty distributions for the global models, local models, power-related parameters, and initial and boundary conditions appear in Tables 1, 2, 3, and 4, respectively, of APP-GW-GLE-026. These distributions are consistent with Tables 1-7, 1-8, 1-10, and 1-11, respectively, of WCAP-16009-P-A. The blowdown heatup heat transfer multiplier was modified to correct modeling inconsistencies and input errors in the WCOBRA/TRAC LOFT deck, as described in Section 15.2.6.5.2.1 of this report.

The initial reactor core power of less than 1.01x3400 MWt assumed a calorimetric uncertainty of 1 percent. This is consistent with DCD Table 15.0-2, which lists the initial thermal power output assumed for the LBLOCA analysis as 3,434 MWt. As stated in the staff safety evaluation in Section 15.1.0.3.1.1 of this report, the applicant will revise DCD Section 15.0.15 to include COL Information Item 15.0.15.1, which requires the COL holder to calculate the primary power calorimetric uncertainty before fuel load to confirm that the safety analysis primary power calorimetric uncertainty bounds the calculated value. In addition, DCD Table 1.8-2 will identify this as COL Item 15.0-1 in accordance with CI-SRP15.0-SRSB-02. Therefore, the staff finds the use of 1 percent initial core power uncertainty acceptable.

The ranges of the parameters were compared to the TS limiting conditions for operation. The accumulator water temperature is ranged based on AP1000 expected values from 50 degrees F to 120 degrees F, which is consistent with the TS 3.6.5, "Containment Air Temperature," limiting condition for operation for an operable containment.

The accumulator pressure range between 670.0 psia and 765.8 psia was inconsistent with TS 3.5.1, SR 3.5.1.3, which specifies the accumulator pressure range between 637 and 769 pounds-force per square inch gauge (psig). In addition, Table 15.6.5-4 did not include the accumulator water volume range. In its response to RAI-SRP15.6.5-SRSB-13 (ADAMS

Accession Number ML083640473), Westinghouse indicated that the incorrect accumulator pressure and water volume ranges were assumed in the AP1000 BELOCA ASTRUM analysis, and a reanalysis of the top 10 HOTSPOT PCT cases from the ASTRUM run set was performed to evaluate the impact of the incorrect pressure and water volume ranges. Maintaining the seed values used in the original ASTRUM analyses, the accumulator pressures and liquid volumes for the top 10 HOTSPOT cases were determined based on the original sampling for each run and the revised ranges. The evaluation showed that the PCT, MLO, and CWO results reported in APP-GW-GLE-026 remained valid for the revised TS ranges. Westinghouse assessed a 0-degree PCT penalty for this set of closely related errors. It must be noted that Figure 27, "HOTSPOT PCT Versus Effective Break Area Scatter Plot for All 124 Cases," in APP-GW-GLE-026 would likely show some changes in the PCT values for some cases if the complete set of 124 runs were to be reanalyzed with the correct accumulator pressure and water volume ranges. However, the overall conclusions for the limiting break would still be applicable.

Westinghouse also committed to update Table 15.6.5-4 in Revision 17 of the DCD to make the accumulator pressure range and the accumulator water volume ranges consistent with TS 3.5.1, SR 3.5.1.3 and SR 3.5.1.2, respectively. This commitment is considered to be part of CI-SRP15.6.5-SRSB-01.

The percentage of steam generator tube plugging is based on the bounding case (10 percent percent), and DCD Table 5.1-3, "Thermal-Hydraulic Parameters (Nominal)," provides the nominal RCS thermal-hydraulic parameters for this case. The average RCS temperature (T_{AVE}) listed in Table 15.6.5-4 is consistent with DCD Table 5.1-3 for this level of plugging. The hot assembly location, under a support column, is bounding based on the design of the reactor vessel upper core region internals. The pressurizer location, in the intact loop, is bounding based on AP1000 WCOBRA/TRAC sensitivity calculations.

The single-failure assumption is the failure of one CMT isolation valve to open. During a large-break LOCA, the CMT is actuated by the "S" signal, but the CMT flow is shut off upon accumulator actuation and resumes at the end of accumulator injection. This limiting single failure affects the safety injection flow delivery when the CMT begins to inject toward the end of accumulator injection.

One of the plant parameter assumptions is that the offsite power remains available during the LOCA transient. In its response to RAI-SRP15.6.5-SRSB-14 (ADAMS Accession Number ML083590305), Westinghouse provided an offsite power availability/unavailability sensitivity analysis using the AP1000 BELOCA model. It confirmed that the offsite-power-available assumption remains limiting in the AP1000 ASTRUM analysis, because of the effect of RCP operation on the downflow cooling during the blowdown period. With loss of offsite power, the RCPs trip coincident with the break and begin to coast down. With offsite power available, the RCPs continue to run until they are automatically tripped. As a result, more fluid flows from the upper plenum into the hot leg in the broken loop, and less fluid reverses flow from the hot leg of the intact loop into the upper plenum. Therefore, less fluid in the upper plenum is available for blowdown cooling. In the case of a loss of offsite power, the liquid downflow into the hot assembly was observed during blowdown; however, there was no observed downflow of liquid into the hot assembly at the upper core plate with offsite power available. The downflow cooling in the case of a loss of offsite power results in increased blowdown cooling and lower PCT.

The containment backpressure is specified at a bounding minimum value, consistent with the WCAP-16009-P-A methodology. The containment pressure is specified at the break location as

an input table. The AP1000 ASTRUM analysis followed the approved WCAP-16009-P-A methodology for determining the conservative containment backpressure approved for standard Westinghouse PWRs. The reference transient was used to establish the containment pressure response that was applied as a boundary condition in the uncertainty analysis calculations. The inputs to the containment pressure calculation were biased to obtain a conservative (low) pressure transient.

The peak linear heat rate, expressed in terms of total heat flux hot channel factor, F_Q , was compared to AP1000 DCD Table 4.3-2, "Nuclear Design Parameters (First Cycle)," and was found to be consistent with the design data. The hot rod assembly power, expressed in terms of nuclear enthalpy rise hot channel factor, $F_{\Delta H}$, was increased from 1.65 to 1.75. In its response to RAI-SRP15.6.5-SRSB-17 (ADAMS Accession Number ML083590305), Westinghouse clarified that it increased $F_{\Delta H}$ in the AP1000 ASTRUM analysis in order to provide increased margin for the core design. The hot assembly average integrated power (P_{HA}) was correspondingly increased to 1.683 from 1.586. The hot assembly average integrated power is 4 percent lower than the hot rod integrated power, consistent with the standard value applied in the approved ASTRUM, WCAP-16009-P-A. DCD Figure 15.6.4A-13 shows the axial power distribution in terms of normalized power integrals in the bottom third of the core and middle third of the core, which is consistent with the treatment in WCAP-16009-P-A.

Only cold leg breaks were analyzed because the hot leg break location was found to be nonlimiting for the BELOCA methodology. ASTRUM explicitly accounts for the effect of break type, and various break types and sizes, such as double-ended cold leg guillotine (DECLG) and split breaks, are assumed to have an equal chance of being sampled. The break size and type were sampled consistent with this methodology. Westinghouse addressed the limiting break type analysis in its response to RAI-SRP15.6.5-SRSB-14 (ADAMS Accession Number ML083590305). Before performing the detailed ASTRUM uncertainty analyses, confirmatory calculations were performed to identify the limiting settings for some of the major plant parameters. The results of these calculations were used to define the reference transient case. Westinghouse found that the important thermal-hydraulic characteristics of the AP1000 during a large-break LOCA were consistent with those observed in conventional Westinghouse three-loop plant analyses. The AP600 large-break LOCA phenomena identification ranking table indicated that the only high-ranking area of difference between the AP600 and a standard three-loop plant with respect to a large-break LOCA was the delivery of emergency core cooling water through the AP600 DVI lines (see the WCAP-14171, Revision 2, and WCAP-15644, Revision 2, WCOBRA/TRAC code applicability reports for the AP600 and the AP1000, respectively). Westinghouse's experience with BELOCA analyses has shown that, for three-loop plants, either a split break or a double-ended guillotine break may be limiting. The reference transient configuration was determined from simulations of nominal DECLG breaks consistent with the approved ASTRUM for conventional three-loop plants (WCAP-16009-P-A).

For Revision 15 of the DCD, the limiting large-break LOCA analyzed with the CQD methodology was determined to be a DECLG break. In Revision 17, the combination of uncertainty parameters sampled in the ASTRUM analysis resulted in the limiting break being a split break, which is the limiting case for both PCT and MLO. Figure 27 of APP-GW-GLE-026 depicts this result and provides the PCT scatter plot showing the impact of the effective break area on the HOTSPOT PCT analysis.

DCD Section 15.6.5.4A.6 describes the limiting PCT/MLO split break case from the AP1000 ASTRUM analysis with the results shown in Figures 15.6.5.4A-1 through 15.6.5.4A-12. Figure 15.6.5.4A-2 shows that the HOTSPOT PCT occurs during the reflood phase. In its

response to RAI-SRP15.6.5-SRSB-14 (ADAMS Accession Number ML083590305), Westinghouse confirmed that the AP1000 remains reflood limited and that the design changes did not result in the limiting PCT moving to the blowdown time period. Westinghouse inspected the 124 ASTRUM calculations. In 100 runs, the PCT occurred during reflood. Westinghouse then inspected the characteristics of the 24 runs in which the PCT occurred during blowdown. These 24 runs were significantly nonlimiting, with the WCOBRA/TRAC PCTs less than 1200 degrees F. The effective break areas sampled for 21 of these calculations were less than 1.0 times the cold leg area, and one was a split break with an effective break area greater than 2.0. The peaking factor and power shape parameters sampled in the remaining two cases contributed to the low PCTs in these calculations. Overall, the early PCT peak in these 24 runs was attributed to the run-specific combinations of sampled parameters, particularly the sampled break areas for the majority of the runs. Westinghouse determined that the ASTRUM runs in which the calculated PCT peaks occurred early in the transient resulted from the combination of sampled parameters for those runs and not the AP1000 design changes.

DCD Table 15.6.5-6, "Best-Estimate Large-Break Sequence of Events for the Limiting PCT/MLO Case," provides the time line for the sequence of events for the limiting BELOCA. The containment high-2 pressure (8 psig) is assumed to occur 2.2 seconds after the initiation of the break because the massive size of the break causes an immediate, rapid pressurization of the containment. This assumption of 2.2 seconds is consistent with the time delay for the "S" signal on high-2 containment pressure for a large-break LOCA specified in DCD Table 15.0-4a. In its response to RAI-SRP15.6.5-SRSB-15 (ADAMS Accession Number ML083640473), Westinghouse addressed the acceptability of the assumed containment response to the BELOCA spectrum. The 2.2-second assumption overestimates the time to reach the containment pressure high-2 setpoint for a nominal DECLG large-break LOCA. Westinghouse calculations with assumptions biased to obtain a conservatively low pressure transient show the containment pressure to be more than 24 psia, as compared to the 22.7 psia (8 psig) high-2 setpoint, at 2.2 seconds after the break. While 2.2 seconds may not allow enough time to reach the high-2 containment pressure setpoint for the smallest breaks sampled as part of ASTRUM because of the reduced mass and energy release, these smaller break sizes were found to be nonlimiting for the AP1000 as shown in APP-GW-GLE-026, Figure 27.

DCD Section 15.6.5.4A.5 states that local and core-wide cladding oxidation values have been determined using the methodology approved in WCAP-16009-P-A. The detailed CWO calculation procedure described in Section 11.6-2 of WCAP-16009-P-A would require a series of WCOBRA/TRAC runs based on the oxidation value from the limiting hot assembly rod case, but with varying power levels for the other runs in the series to account for lower power assemblies in other core regions. For the AP1000, Westinghouse chose to apply the limiting case hot assembly rod oxidation value to the entire reactor core. In its response to RAI-SRP15.6.5-SRSB-18 (ADAMS Accession Number ML083640469), Westinghouse clarified that the method used to evaluate the limiting CWO is a conservative approach and in accordance with the procedure used for standard ASTRUM analyses. The results of the 124 cases were ranked by the WCOBRA/TRAC-calculated hot assembly average volumetric oxidation from highest to lowest. The case with the maximum hot assembly average volumetric oxidation was examined. In the AP1000 ASTRUM analysis, this was case "run015" with a hot assembly oxidation of 0.2 percent. Since this oxidation was substantially lower than the regulatory CWO limit of 1 percent, a detailed calculation of the CWO was not necessary for the AP1000. By definition, the CWO fraction is less than the hot assembly average volumetric oxidation because the many lower power assemblies present in the reactor core will have lower average volumetric oxidation than the hot assembly. The application of the limiting hot assembly rod oxidation value to the entire core is conservative and acceptable.

15.2.6.5.2.4 Summary of the Best Estimate Large Break Loss-of-Coolant Accident Analysis Results

The results of the AP1000 BELOCA ASTRUM analyses, based on the 124 WCOBRA/TRAC and HOTSPOT runs, identify the limiting PCT/MLO split break. DCD Table 15.6.5-8 summarizes the AP1000 ASTRUM BELOCA results and shows the calculated 95th percentile PCT to be 1837 degrees F, the MLO to be 2.25 percent, and the limiting CWO to be 0.2 percent. Table 15.6.5-8 also indicates that the core remains coolable and the core remains cool in the long term. The core coolable geometry is generally satisfied when calculated PCT and MLO are within the associated acceptance criteria. The post-LOCA long-term core cooling, as described in DCD Section 15.6.5.4C, "Post-LOCA Long-Term Cooling," is not affected by the application of ASTRUM and continues to show compliance with 10 CFR 50.46 acceptance criteria. Therefore, the BELOCA analysis results show a high level of probability that the following criteria of 10 CFR 50.46 will be met:

- The calculated PCT will not exceed 2,200 degrees F.
- The calculated maximum cladding oxidation will not exceed 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, the core temperature will be maintained at an acceptably low value, and the decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

The calculated results for the AP1000 BELOCA meet the acceptance criteria of 10 CFR 50.46 and are therefore acceptable.

15.2.6.5.2.5 Conclusion

Based on the above evaluation, the staff concludes, upon the successful resolution of CI-SRP15.6.5-SRSB-01, that the BELOCA analysis using the WCOBRA/TRAC code and the ASTRUM statistical uncertainty treatment methodology described in Revision 17 of the DCD demonstrates compliance with the acceptance criteria of 10 CFR 50.46 and therefore is acceptable. Sections 15.2.6.5.2.2.3 and 15.2.6.5.2.3.2 of this report describe the details of CI-SRP15.6.5-SRSB-01, which is summarized below:

- The applicant will revise DCD Section 15.6.5.4A.5, "Accumulator Injection Flow for 95th Percentile Estimator PCT/MLO Case," to incorporate the result of the sensitivity studies for the isolation of CMT and PRHR.
- The applicant will revise DCD Table 15.6.5-4 to make the RCS average temperature, accumulator temperature, pressure, and water volume range consistent with TS values.

15.2.7 Post-Loss-of-Coolant Accident Long-Term Cooling (DCD Tier 2, Section 15.6.5.4C)

DCD Section 15.6.5.4C did not change; however, the NRC is reviewing the long-term cooling analyses and supporting documentation submitted for resolution of COL Item 6.3.8.2 related to Generic Safety Issue-191 (GSI-191), "Assessment of Debris Accumulation on PWR Sumps," for the AP1000, together with the GSI-191 issues addressed in Section 6.2.1.8, "Adequacy of IRWST and Containment Recirculation Screen Performance." For consistency, the staff's evaluation and conclusion regarding post-LOCA long-term cooling for the AP1000 appear with the GSI-191 discussion in Section 6.2.1.8 of this report.

15.3 Radiological Consequences of Accidents

In DCD Tier 2, Chapter 15, the applicant performed radiological consequence assessments of the following seven design-basis accidents (DBAs), using the hypothetical set of atmospheric dispersion factors (χ/Q values) provided in DCD Tier 1, Table 5.0-1, "Site Parameters"; DCD Tier 2, Table 2-1, "Site Parameters"; DCD Tier 2, Table 15A-5, "Offsite Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis"; and Table 15A-6, "Control Room Atmospheric Dispersion Factors (X/Q) for Accident Dose Analysis":

- (1) LOCA
- (2) fuel-handling accident (FHA)
- (3) main steamline break accident outside of containment
- (4) RCP shaft seizure accident
- (5) failure of small lines carrying primary coolant outside containment
- (6) rod cluster assembly ejection accident, and
- (7) steam generator tube rupture accident

The applicant concluded in DCD Tier 2, Revision 15, 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 specified in 10 CFR 50.34(a)(1) and the control room operator dose criterion specified in GDC 19, "Control Room," of Appendix A to 10 CFR Part 50. In NUREG-1793, the staff performed independent radiological consequence analyses for all of the DBAs listed above and verified the applicant's assessments.

In September 2008, the applicant submitted Revision 17 of the DCD, as a part of its application to amend the AP1000 design certification rule in Appendix D to 10 CFR Part 52. In Revision 17, the applicant requested the following four standard changes to the Chapter 15 of the certified AP1000 DCD, Revision 15:

- (1) increase in the assumed decay time in the FHA dose analysis from 24 hours to 48 hours to provide increased radioactive decay of short-lived fission products before the handling of irradiated fuel assemblies
- (2) increase in the aerosol removal duration in the containment from 15.5 hours to 24 hours from the initiation of a DBA

- (3) revisions to some of the hypothetical offsite and control room χ/Q values, and
- (4) changes to the main control room emergency habitability system operation

The applicant proposed these changes in accordance with the change criterion in 10 CFR 52.63(a)(1)(vii) in that the changes contribute to increased standardization of the certification information.

Subsequent to the submittal of the AP1000 DCD Revision 17, on November 3, 2008, the applicant proposed to add a first-of-a-kind passive control room air filtration line to the main control room emergency habitability system (VES). This design change is intended to allow the VES to meet the dose acceptance criterion specified in GDC 19 with an allowable control room unfiltered air inleakage of 15 cubic feet per minute (cfm), which includes 5 cfm for ingress/egress. Section 6.4 of this SE (Supplement 2 to NUREG-1793) gives the staff's evaluation of the main control room VES, including the proposed design changes.

These changes will alter the calculated radiological doses in the control room for the above DBAs analyzed in the certified AP1000 DCD, Revision 15, and will revise some of the χ/Q values listed as site parameters in the following tables:

- DCD Tier 1, Table 5.0-1, "Site Parameters"
- DCD Tier 2, Table 2-1, "Site Parameters"
- DCD Tier 2, Table 15A-5, "Offsite Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis"
- DCD Tier 2, Table 15A-6, "Control Room Atmospheric Dispersion Factors (χ/Q) for Accident Dose Analysis"

Revision 17 of the DCD, Section 2.3.4, evaluates the proposed changes to the hypothetical short-term accident χ/Q s, and Table 2.3.4-1 lists the accident χ/Q values used as site parameters.

15.3.1 Evaluation

15.3.1.1 Fuel-Handling Accident Decay Time Increase

In Revision 15 of the DCD, Section 15.7.4, "Fuel Handling Accident," the applicant analyzed the radiological consequences of a postulated FHA inside containment and in the fuel-handling area inside the auxiliary building, assuming that a single fuel assembly that has undergone 24 hours of decay time is dropped. This decay time provides for radioactive decay of short-lived fission products to reduce the radiological consequences of a release for the design-basis FHA. Even though the applicant analyzed the FHA using a decay time of 24 hours in Revision 15 of the DCD, the applicant had conservatively specified a 100-hour decay time in Revision 15 of the DCD, Chapter 16, TS 3.9.7, "Decay Time." The longer decay time would result in further reduction of radiological consequences of the FHA than the applicant calculated in the FHA radiological consequences analysis in Revision 15 of the DCD.

As discussed in NUREG-1793, the staff reviewed the applicant's analysis and performed an independent confirmatory radiological analysis for the postulated FHA. The staff's results agree with the applicant's values. Both the applicant's and the staff's results met the relevant dose acceptance criteria at the exclusion area boundary, low-population zone (LPZ), and control room. Based on its review, the staff finds the applicant's analysis of the FHA to be acceptable and that the dose criteria in 10 CFR 50.34(a)(1) and GDC 19 are met.

In Revision 17 of the DCD, the applicant proposed to change the minimum decay time to 48 hours from the 24 hours assumed in Revision 15 of the DCD and in NUREG-1793. This change will provide additional radioactive decay of short-lived fission products, which would result in a lesser amount of radioactive material available for release and therefore lower dose results for the radiological consequence assessment for the postulated FHA. Revision 17 of the DCD, Section 15.7.4, discussed the revised FHA analysis assuming a 48-hour decay time. The staff verified that the only changes to the analyses were related to the change in decay time and the changes in the hypothetical χ/Q values discussed below in Section 15.3.1.3.

The staff finds that the change in the FHA radiological consequence analysis ensures that the decay time assumption in the analysis is consistent with the minimum decay time value proposed for the AP1000 TS and results in lower control room and offsite doses than previously found acceptable for Revision 15 of the DCD. Therefore, the staff finds that the proposed change to the TS decay time is acceptable with respect to the radiological consequences of DBAs. The resulting doses off site and in the control room continue to meet the dose acceptance criteria in Regulatory Guide 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000, and SRP Section 15.0.3 for the design certification review. The applicant's revised FHA dose analyses show that the dose criteria in 10 CFR 50.34(a)(1), 10 CFR 52.47(a)(2)(iv), and GDC 19 would be met with the change in minimum decay time before fuel movement. The applicant revised the decay time requirement in Revision 17 of the DCD, TS 3.9.7, to 48 hours from 100 hours to be consistent with the revised FHA analysis. The staff finds that the proposed change to the TS is supported by the FHA analysis and therefore is acceptable.

15.3.1.2 Aerosol Removal Duration in Containment

In Table 15.6.5-3, "Radiological Consequences of a Loss-of-Coolant Accident with Core Melt," of Revisions 15 and 17 of the DCD, the applicant provided the LPZ doses as 23.8 and 23.4 rem TEDE, respectively. In RAI-SRP15.6.5-RSAC-01 (ADAMS Accession Number ML083590302), the staff asked the applicant to explain the discrepancy in these two LPZ doses, since none of the analysis assumptions given in Revision 15 appear to have changed in Revision 17. In its response to this RAI (ADAMS Accession Number ML083590302), the applicant stated that the analyses in Revision 17 of the DCD took full credit for aerosol removal for the first 24 hours (rather than for 15.5 hours as in Revision 15), using the removal coefficients presented in Appendix 15B to Revision 15 of the DCD. This resulted in a small decrease in the LPZ dose to 23.4 rem TEDE. The staff had previously found the aerosol removal coefficients credited for the first 24 hours in Appendix 15B to Revision 15 of the DCD to be acceptable. Revision 17 did not change any of the aerosol removal coefficients in Appendix 15B. The staff finds that the response to RAI-SRP15.6.5-RSAC-01 is acceptable; therefore, this RAI is closed and the changes to the assumed duration of the aerosol removal in containment are acceptable.

15.3.1.3 Offsite and Control Room χ/Q Values

15.3.1.3.1 Control Room χ/Q Values

In Revision 16 of the DCD, the applicant increased the control room χ/Q values from those specified in Revision 15. In Revision 17, the applicant proposed a change in the control room isolation logic so that the increased χ/Q values in Revision 16 could be maintained, with two exceptions.

The χ/Q values for the heating, ventilation, and air conditioning (HVAC) intake at the ground level containment release points in the 2–8-hour and 8–24-hour intervals are reduced for the LOCA with the nuclear island nonradioactive ventilation system in supplemental filtration mode.

In addition, in Revision 17 of the DCD, the applicant: (1) revised the control room χ/Q values for plant vent/passive containment system air diffuser and ground level containment releases to the control room HVAC intake and Annex Building door, and (2) added new control room χ/Q values for condenser air removal stack releases to the HVAC intake and Annex Building door. Revision 17 of the DCD, Table 2.3.4-1, lists these revisions. Section 2.3.4 of this report provides the staff's evaluation and acceptance of these revisions.

15.3.1.3.2 Offsite χ/Q Values

Revision 17 of the DCD does not change the offsite χ/Q values provided in Revision 15.

15.3.1.4 Offsite and Control Room Doses

15.3.1.4.1 Offsite Doses

The applicant has not changed the radiological doses calculated for the postulated LOCA from those in Revision 15 of the DCD, except for the small decrease for the LPZ dose evaluated in Section 15.3.1.2 of this report. For the radiological consequences of DBAs other than the LOCA, the applicant used the higher and more restrictive χ/Q values provided in Revision 17 of the DCD (compared to those χ/Q values used in Revision 15), while still meeting the dose acceptance criteria. The staff verified and confirmed the applicant's revised offsite doses for the DBAs other than the LOCA. The radiological consequence analyses for the five remaining DBAs (other than the LOCA and FHA as discussed above) have not changed in Revision 17 of the DCD from those described in Revision 15.

Therefore, the DBA analyses in Revision 17 of the DCD used two sets of χ/Q values, one for the LOCA and the other for the DBAs other than the LOCA, although Table 15.A-5 shows only one set of χ/Q values used or LOCA. In its response to RAI-SRP15.6.5-RSAC-02 (ADAMS Accession Number ML083590302), the applicant stated that Table 15A-5 of a future DCD revision will include these two sets of χ/Q values.

Subsequently, the applicant indicated during the November 3, 2008, meeting with the staff that it will use the same offsite χ/Q values for all DBAs in future DCD revisions to avoid having two sets of χ/Q values, one for the LOCA and another for the DBAs other than the LOCA. As a result, the next DCD Revision 18 will include revised offsite doses for all DBAs except the LOCA. This is CI-SRP15.0.3-RSAC-15, for forthcoming AP1000 DCD Revision 18.

15.3.1.4.2 Control Room Doses

In Revision 17 of the DCD, the applicant also proposed to modify the control room isolation to include switching to the main control room VES on the pressurizer low-pressure signal. In the

LOCA dose analysis, this modification results in actuation of the VES by the time core activity releases start at 10 minutes with leak-before-break approval. In addition, the applicant proposed an effective unfiltered air leakage assumption of 1.5 cfm into the control room based on a total leakage of 5 cfm, with credit taken for purging of the vestibule door volume and the incomplete mixing of the vestibule and control room volumes with outside air following ingress/egress.

During the November 3, 2008, meeting with the staff, the applicant presented the addition of a new passive control room air filtration line to the VES. This design change is intended to allow the VES to meet the dose acceptance criterion specified in GDC 19 with an allowable unfiltered air leakage of 15 cfm instead of the assumption of 5 cfm total unfiltered leakage as discussed above.

To evaluate the revised control room passive filtration design, the staff performed an independent radiological consequence dose calculation for the control room and audited the applicant's dose calculations with the new passive control room air filtration line to the VES. Based on this review of the applicant's analyses, the staff verified that the control room habitability system and the technical support center design in the AP1000 DCD, revision 17, as modified in RAI response dated May 24, 2010, meet the dose acceptance criteria specified in GDC 19 and SRP Section 15.0.3.

The staff previously stated that upon completion of the review and acceptance of this new passive control room air filtration line to the VES, the staff will complete its independent radiological consequence dose calculations for the control room to verify that the control room habitability system and the technical support center (TSC) designed in the AP1000 DCD meet the dose acceptance criteria specified in GDC 19 and SRP 15.0.3. This was Open Item OI-SRP15.3-1-RSAC-01. With completion of staff's review and an independent dose calculation as discussed above, Open Item OI-SRP15.3-1-RSAC-01 is closed.

Section 6.4 of this SE gives the staff's evaluation and acceptance of the new passive control room air filtration line. This design change proposed and accepted by the staff should be incorporated into forthcoming AP1000 DCD Revision 18. This is identified in the Chapter 6, Section 6.4, SE as **CI-SRP6.4-SPCV-15**.

15.3.2 Conclusion

Based on the above evaluation, the staff concludes that Revision 17 of the DCD provides reasonable assurance that the radiological consequences resulting from any of the DBAs will fall within the dose acceptance criteria specified in SRP 15.0.3, 10 CFR 52.47(a)(2), GDC 19, and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50 for the exclusion area boundary, LPZ, and control room. The staff further concludes that Revision 17 of the DCD also provides reasonable assurance that the AP1000 technical support center will meet radiological consequences criteria specified in NUREG-0737, Supplement Number 1 and in SRP Section 15.0.3.