4. REACTOR

4.1 Introduction

In Revision 17 of the AP1000 design control document (DCD), Westinghouse Electric Company, LLC (Westinghouse or the applicant) proposed changes related to the reactor core and fuel design. In a letter dated October 31, 2006, to the U.S. Nuclear Regulatory Commission (NRC), Westinghouse submitted Westinghouse Commercial Atomic Power (WCAP)-16652-NP, Revision 0, APP-GW-GLR-059, "AP1000 Core & Fuel Design Technical Report" (technical report (TR)-18), to justify the proposed changes. Various sections of DCD Chapter 4, Revision 17, incorporate the majority of these proposed changes. The NRC staff's evaluation of the proposed changes for each section of Chapter 4 of the AP1000 DCD is addressed in the corresponding section of this safety evaluation. This particular section describes the staff's evaluation of the proposed changes to DCD Section 4.1, "Summary Description."

In its letter of May 26, 2007, regarding its application to amend the AP1000 design certification (DC) rule, Westinghouse referred to the criterion of Title 10 of the *Code of Federal Regulations* (10 CFR) 52.63(a)(1)(vii)), "Finality of standard design certifications" and noted that these proposed changes contribute to increased standardization of the certification information.

4.1.1 Evaluation

General Design Criterion (GDC) 10, "Reactor Design," of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic licensing of production and utilization facilities," requires that the reactor core and associated coolant, control, and protection systems be designed to assure that the specified acceptable fuel design limits (SAFDLs) are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOOs). GDC 13, "Instrumentation and Control," requires that instrumentation be provided to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions. GDC 26, "Reactivity Control System Redundancy and Capability," requires that two independent reactivity control systems of different design principles be provided to ensure that SAFDLs are not exceeded and that the reactor core can be held subcritical under cold conditions. GDC 27, "Combined Reactivity Control Systems Capability," requires that the reactivity control systems be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system (ECCS), of reliably controlling reactivity changes to ensure that, under postulated accident conditions and with appropriate margins for stuck rods, the capability to cool the core is maintained. GDC 28, "Reactivity Limits," requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents will not result in damage to the reactor coolant pressure boundary (RCPB) or sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. Various sections of Chapter 4 of the AP1000 DCD pertaining to the reactor core, fuel design, nuclear design, thermal-hydraulic design, and reactivity control system design describe compliance with these GDC.

DCD Section 4.1, "Summary Description," describes the AP1000 reactor core and fuel design. The staff reviewed the proposed changes to DCD Section 4.1 and other sections in Chapter 4 for compliance with these GDC. In DCD Revision 17, the applicant proposed the following

changes to DCD Tier 2, Section 4.1, related to the AP1000 reactor core and fuel design: (1) clarification of the fuel protective grid design and modification of the top nozzle design nomenclature; (2) modification of the gray rod cluster assembly (GRCA) design; (3) modification of the values of several parameters in DCD Section 4.1.1 and Table 4.1-1 for consistency; and (4) addition of four neutron panels on the thermal shield.

The current DCD Section 4.1 states that the top and bottom grids of the fuel assembly design do not contain mixing vanes. In DCD Revision 17, the applicant revised this sentence to state that the top and bottom grids and the protective grid do not contain mixing vanes. This clarification specifically states that the protective grid also does not contain mixing vanes. In DCD Table 4.1.1, the applicant revised the number of grids per fuel assembly to specifically identify one protective grid. This is a clarification of the design, not a design change, and is, therefore, acceptable.

In DCD Revision 17, the applicant proposed to change "integral clamp top nozzle (ICTN)" to "Westinghouse integral nozzle (WIN)" in the fuel assembly design. In TR-18, the applicant stated that the change from a reconstitutable ICTN to WIN in the fuel assembly is a nomenclature change. The WIN design is a proven enhancement to the ICTN design and is currently in use in the Westinghouse fleet. Both the ICTN and WIN designs eliminate the need for the top nozzle spring screws and spacer clamps. There is no significant design difference between the ICTN and WIN. The applicant described the design refinements of WIN, as compared to the ICTN, in a letter dated April 19, 2004, regarding the fuel criterion evaluation process (FCEP) notification of the WIN design (proprietary and nonproprietary) (Agencywide Documents Access and Management System (ADAMS) Accession Nos. ML041120331 and ML041120332). The staff finds that the applicant developed the WIN design using the fuel criteria evaluation process described in WCAP-12488. Therefore, the change as presented in DCD Revision 17 is acceptable.

In DCD Revision 17, the applicant proposed to change the GRCA design from 4 rodlets (out of 24) containing silver-indium-cadmium (Ag-In-Cd) to 12 rodlets containing Ag-In-Cd with reduced diameter, as described in DCD Section 4.2.2.3.2. The applicant revised DCD Table 4.1-1 to reflect this change. As discussed in Section 4.2 of this report, the staff finds the change to the GRCA design acceptable.

In DCD Section 4.1.1, the applicant changed the calculated core average power limit in the principal design requirement from 18.73 kilowatts per meter (kW/m) (5.71 kilowatts per foot (kW/ft)) to18.76 kW/m (5.718 kW/ft). The value of 18.73 kW/m (5.71 kW/ft) is a truncation of the actual value of 18.76 kW/m (5.718 kW/ft). DCD Tables 4.1-1 and 4.3-2 round the core average linear power to 18.77 kW/m (5.72 kW/ft). Therefore, the staff finds the change in Section 4.1.1 to 18.76 kW/m (5.718 kW/ft) acceptable. The applicant also revised several parameters in DCD Table 4.1-1. The effective reactor coolant flow area of heat transfer, the average velocity along the fuel rods, and the average mass velocity were revised slightly to be consistent with DCD Table 4.4-1 and the revised definition of core flow area and the core bypass flow described in DCD Section 4.4. Section 4.4 of this report addresses the revision to the core flow area and bypass.

The applicant revised DCD Table 4.1-1 to include four neutron panels on the thermal shield. The applicant made this revision to be consistent with the reactor internals design changes, which call for the addition of four neutron panels to reduce the reactor vessel fluence at the circumferential locations that have the highest fluence values, along with the addition of a flow skirt in the lower reactor vessel head, as described in WCAP-16716-NP, Revision 2, "AP1000

Reactor Internals Design Changes" (TR-29). Since these changes could affect the core inlet flow distribution and the flow area and flow resistance in the reactor vessel downcomer and lower plenum, the staff requested that the applicant provide an evaluation of the impacts of these proposed changes on the analysis results of each of the transients and accidents described in DCD Chapter 15, "Accident Analyses."

In response to request for additional information (RAI)-TR29-SRSB-01, the applicant presented the results of the evaluation and safety analyses of the reactor vessel internals design changes. This evaluation also included the pressurizer changes described in TR-36, APP-GW-GLR-016, Revision 0, "AP1000 Pressurizer Design." (In its letter of July 18, 2008, in response to RAI-SRP10.3-SBPA-02, the applicant also stated that the evaluation of the limiting Chapter 15 event analyses provided in RAI-TR29-SRSB-01 included the revised main steam safety valve setpoints and capabilities.) The applicant concluded that these reactor internals changes have minimal impact on the fluid volume, metal masses, pressure drop through the reactor vessel, and the design reactor coolant system (RCS) flow rates used in the safety analyses. The applicant performed the evaluation or analyses on the limiting event of each of the event categories discussed in DCD Chapter 15. The analysis results for these events demonstrate that the applicable acceptance criteria for each event are met or that the existing analysis is bounding.

In its June 30, 2008, letter, the applicant provided APP-GW-GLE-026, Revision 0, "Application of ASTRUM Methodology for Best Estimate Large Break Loss of Coolant Accident Analysis for AP1000." Subsequently, the applicant, in a letter dated February 3, 2009, submitted APP-GW-GLE-026, Revision 1. This TR describes a reanalysis of a large-break loss-of-coolant accident (LOCA) using the NRC-approved uncertainty treatment method, Automatic Statistical Treatment of Uncertainty Method (ASTRUM). This large-break LOCA reanalysis includes the reactor vessel internals design changes discussed above. The results of this large-break LOCA analysis with the design changes to date show compliance with the acceptance criteria of 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors." In the review of large-break LOCAs, Section 15.2.6.5 of this report addresses the staff evaluation of this TR.

DCD Section 4.1.1, "Principal Design Requirements," lists the criteria, or principal design requirements, that must be met by the mechanical design and physical arrangement of the reactor components, together with the corrective actions of the reactor control, protection, and emergency cooling systems. These principal design criteria are designated Tier 2* information, which indicates that staff approval will be required before implementation of any change to this information. Various sections of Chapter 4 of this report describe other Tier 2* information related to the reference core design. In Section 4.1 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," the staff states that the following sections in DCD Tier 2, Chapter 4, include Tier 2* information:

- WCAP-12488-A, "Westinghouse Fuel Criteria Evaluation Process," issued October 1994
- Principal Design Requirements
- Maximum Fuel Rod Average Burnup of 62,000 Megawatt-Days per Metric Ton of Uranium (MWDF/MTU)
- Table 4.3-1, Reactor Core Description (First Cycle)

- Table 4.3-2, Nuclear Design Parameters (First Cycle)

- Table 4.3-3, Reactivity Requirements for Rod Cluster Control Assemblies

In DCD Revision 16, the applicant proposed to reclassify all Tier 2* information throughout Chapter 4 as Tier 2 information to allow future changes to this information to be implemented without prior NRC approval. However, in response to RAI-SRP4.2-SRSB-01, the applicant stated that it would withdraw the request that Tier 2* items be reassigned to Tier 2 status in Chapter 4. Therefore, all Tier 2* items in DCD Chapter 4, including the principal design requirements, remain Tier 2* items. In DCD Revision 17, all Tier 2* information currently identified in DCD Revision 15 is restored to Tier 2*. In DCD "Introduction," Table 1-1, "Index of AP1000 Tier 2 Information Requiring NRC Approval for Change," these Tier 2* items, which were included in DCD Revision 15 but deleted in Revision 16, have been restored. The staff finds this acceptable.

4.1.2 Conclusion

Based on this evaluation, the staff concludes that the changes to DCD Section 4.1 continue to meet the requirement of GDCs 10, 13, 26 and 28 as described above. In addition, the results of the large break LOCA analysis with these design changes comply with the acceptance criteria of 10 CFR 50.46. Therefore, the proposed changes are acceptable.

4.2 Fuel System Design

Revision 17 of the AP1000 DCD includes the following changes to Section 4.2 (as compared to Revision 15): (1) reference changes; (2) densification value; (3) control rod descriptions; (4) burnable absorber rod design; (5) debris protection package description; (6) top nozzle nomenclature; and (7) grid fabrication description.

The staff based its review of the AP1000 fuel design on the information in the DCD and the TRs referenced by the applicant. The review was limited in scope to the changes to DCD Revision 15, as presented in Revision 17. The staff conducted its review in accordance with the guidelines provided in NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 4.2, "Fuel System Design," which prescribes acceptance criteria to ensure that certain requirements of 10 CFR Part 50 are met. In particular, the AP1000 fuel design must meet the following GDC:

- GDC 10
- GDC 27
- GDC 35, "Emergency Core Cooling"

The fuel design must also meet the requirements of 10 CFR Part 100, "Reactor site criteria." Thus, in reviewing the AP1000 fuel system design, the staff's objective was to ensure that the design fulfills the following criteria:

- The fuel system will not be damaged during any condition of normal operation, including the effects of AOOs.

- Fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when required.

- The number of fuel rod failures for postulated accidents is not underestimated.

- Coolability is always maintained.

The term "will not be damaged" means that the fuel rods will not fail, the fuel system's dimensions will remain within operational tolerances, and the functional capabilities will not be reduced below those assumed in the safety analysis. These objectives address GDC 10, and the design limits that accomplish these objectives are referred to as SAFDLs. In a "fuel rod failure," the fuel rod leaks and the first fission product barrier (i.e., the fuel cladding) is breached. The applicant must account for fuel rod failure in the dose analysis for postulated accidents required by 10 CFR Part 100. The radiological dose consequences criteria given in 10 CFR 50.34(a)(1), "Contents of applications; technical information," are referenced in 10 CFR 100.21, "Non-seismic siting criteria." Compliance with dose consequence criteria in 10 CFR 50.34(a)(1), with the site parameters postulated for the design, is discussed in Section 15.3 of this report.

"Coolability," which is sometimes termed "coolable geometry," is the ability of the fuel assembly to retain the geometrical configuration of its rod bundle with adequate coolant channel spacing for removal of residual heat. GDC 27 and GDC 35 specify the general requirements for maintaining control rod insertability and core coolability. In addition, 10 CFR 50.46 establishes specific requirements for the performance of the ECCS following postulated LOCAs. Compliance with 10 CFR 50.46 is discussed in Section 15.2.6.5 of this report.

4.2.1 Evaluation

The applicant made various reference changes in Revision 17. When needed, the applicant included additional references. In other cases, the applicant replaced references to correct erroneous references found in Revision 15. For example, the applicant deleted an erroneous reference for the integral fuel burnable absorber design in Section 4.2.2.1. In RAI-SRP4.2-SRSB-04, the staff requested that the applicant include the correct reference instead of simply removing the incorrect reference. In its response, the applicant committed to add a reference to WCAP-12610-P-A to the conclusion of Section 4.2.2.1. DCD Section 4.2.6 includes this report as Reference 5. The staff has reviewed these changes and finds them acceptable.

Throughout Section 4.2, the applicant revised the description of the control rod assemblies to clarify that the control rod clusters consist of both rod cluster control assemblies (RCCA) and gray rod cluster assemblies (GRCA). The term gray rod refers to clusters of neutron absorber rods with reduced rod worth relative to that of an RCCA. The GRCAs are used in load follow maneuvering to provide a mechanical shim reactivity mechanism to eliminate the need for changes to the concentration of soluble boron. Additionally, the applicant changed the GRCA design. Specifically, the applicant increased the number of rodlets from 4 to 12 and correspondingly reduced the diameter, such that the overall worth of the GRCA is essentially the same, as explained in the applicant's response to RAI-SRP4.2-SRSB-05, but the assembly power profile is more even. The change in the GRCA provides a more distributed absorber material within the assembly. By reducing the diameter of the absorber material, and dispersing the absorber over more rodlets, the reactivity worth of the GRCAs is maintained while lessening the local power perturbation. DCD Revision 17 does not present the impact of these GRCAs on

determining the worst-case scenarios for accident analyses. The applicant's response to RAI-SRP4.2-SRSB-05 indicated that the Chapter 15 analyses remain bounding because the use of the newer GRCA design results in a more uniform rod power throughout the affected assemblies and does not affect the worst-case assumptions used in the accident analyses. The NRC staff found this acceptable based on the RAI response and the nonsafety status of the GRCAs.

The applicant changed the burnable absorber rod design from wet annular burnable absorbers (WABAs) to borosilicate glass. However, the reference for the borosilicate design presented (WCAP-7113) is from 1967 and does not appear to have been previously approved by the NRC.

Additionally, the DCD continues to list the WABA design as an alternative. The staff issued RAI-SRP4.2-SRSB-02, RAI-SRP4.2-SRSB-03, and RAI-SRP4.2-SRSB-07 regarding these items. The applicant's response pointed out previous NRC approval of Reference Safety Analysis Reports (RESARs), that used the same reference (WCAP-7113), covering borosilicate glass RCCAs, but the applicant did not provide a specific RESAR reference. The applicant also referred to operational experience with the borosilicate glass rodlets, including a single failure of a rodlet that became detached and remained in the guide tube. In response to RAI-SRP4.2-SRSB-07, the applicant states that borosilicate burnable absorbers were used previously in a Westinghouse core design with a 14-ft core and a feed fuel enrichment of 4.4 percent. This is very similar to the AP1000 core. The response further explained that there are no changes to rodlet diameter and no changes to boron concentration for the AP1000 design as compared with the previously approved design. Based on the responses to RAI-SRP4.2-SRSB-02, RAI-SRP4.2-SRSB03, and RAI-SRP4.2-SRSB-07, the staff concludes that the use of borosilicate glass burnable absorbers is acceptable.

The applicant provided further description of the debris mitigation package in DCD Sections 4.2.2 and 4.2.2.1. Specifically, the applicant explicitly listed the parts of the debris mitigation package that include a zirconium oxide coating on the bottom section of the fuel cladding.

The applicant changed the top nozzle design in Section 4.2 (and other sections) from "Integral Clamp Top Nozzle (ICTN)" to "Westinghouse Integral Nozzle (WIN)." As described in Section 4.1.1 of this safety evaluation report (SER), the staff found this change to be acceptable.

The applicant further explained the grid fabrication process in Section 4.2.2.2.4. This information did not appear to change the grid design from that presented in DCD Revision 15 and is, therefore, acceptable.

In accordance with TR-119, APP-GW-GLR-119, "AP1000 Design Control Document Chapter 4 Tier 2* Information," the applicant edited DCD Revision 16 to remove Tier 2* designation from various fuel-related items. The applicant argued, in TR-119, that these items would be provided in a future "core reference report" and as such they could be reclassified as Tier 2. In RAI-SRP4.2-SRSB-01, the staff expressed concern that the reclassification could theoretically be interpreted to allow changes to the affected items after the core reference report was approved. In response, the applicant stated that it would withdraw the request for reclassification of all items presented in TR-119 and restore, in DCD Revision 17, all Tier 2* information currently identified in DCD Revision 15. The staff verified that Section 4.2 of Revision 17 includes all of the items designated as Tier 2* in Revision 15. In addition,

Revision 17 for Table 1-1 of the DCD "Introduction" also includes these Tier 2* items, which require NRC approval for change. This staff finds this acceptable.

In Revision 15 of the DCD, Sections 4.2.5, 4.3.4, and 4.4.7 state that combined license (COL) applicants referencing the AP1000 certified design will address changes to the reference design presented in the DCD to the fuel burnable absorber rods, RCCAs, or initial core design. DCD Tier 2, Table 1.8-2, lists these COL applicant actions as COL Information Items 4.2-1, 4.3-1, and 4.4-1. In DCD Revision 17, the applicant proposed to revise Sections 4.2.5, 4.3.4, and 4.4.7.1 to state that: (1) APP-GW-GLR-059 completely addressed the COL information requested in these sections; (2) the DCD incorporated applicable changes; and (3) no additional work is required by the COL applicant to address the COL information requested in this section. In DCD Revision 17, the applicant revised Table 1.8-2 to require no COL applicant action for COL Information Items 4.2-1, 4.3-1, and 4.4-1. These COL information items intend for the COL applicant to provide information regarding the changes to the referenced reactor core design in the AP1000 DC. To increase standardization of the certified design, the applicant submitted APP-GW-GLR-059, which provides information that addresses changes to the reference design of the fuel burnable absorber rods, RCCAs, or initial core design in Revision 15 of DCD. DCD Revision 17 incorporates the applicable changes.

Based on the evaluation discussed in this report, the staff concludes that COL Information Items 4.2-1, 4.3-1, and 4.4-1 associated with DCD Revision 15 have been completed, and Revision 17 of DCD Sections 4.2.5, 4.3.4, and 4.4.7.1 is acceptable. It should be noted that the referenced design of the fuel burnable absorber rods, RCCAs, and initial core design parameters described in DCD Chapter 4 are classified as Tier 2* information, as described in Section 4.1.1. It is likely that future advancements in the design of these items could occur and COL applicants might desire these improved designs. Any change to the Tier 2* information will require prior approval by the NRC.

4.2.2 Conclusion

The staff concludes that the AP1000 fuel system, as defined by the DCD Revision 17 changes to the approved design, has been designed so that: (1) the fuel system will not be damaged as a result of normal operation and AOOs; (2) fuel damage during postulated accidents will not be severe enough to prevent control rod insertion when it is required; and (3) core coolability will always be maintained, even after severe postulated accidents, thereby meeting the related requirements of 10 CFR 50.46; GDC 10, 27, and 35; and 10 CFR Part 50.34.

The applicant provided sufficient information indicating that the changes to the approved design, as detailed in the DCD, meet the guidance provided in NUREG-0800 Section 4.2.

4.3 Nuclear Design

The staff reviewed Section 4.3 in Revision 17 of the AP1000 DCD. The staff conducted its evaluation in accordance with the guidelines provided in NUREG-0800 Section 4.3, "Nuclear Design."

In DCD Revision 17, Section 4.3, "Nuclear Design," the applicant proposed changes to the following areas related to the AP1000 nuclear design: (1) online monitoring of power distribution; (2) gray rod assembly design; (3) criticality design method outside the reactor, including soluble boron credit methodology; (4) deletion of a specific value for the moderator temperature change that accounts for the control system deadband; and (5) typical control bank

worth for the initial cycle. In support of these changes, the applicant submitted APP-GW-GLR-059. The applicant originally proposed in DCD Revision 16 to reclassify the Tier 2* information in DCD Section 4.3 to Tier 2. However, in DCD Revision 17, the applicant rescinded this proposal and restored this information to its original Tier 2* designation.

4.3.1 Evaluation

DCD Tier 2, Section 4.3, "Nuclear Design," presents the design bases for the AP1000 nuclear design. The nuclear design must ensure that the SAFDLs will not be exceeded during normal operation, including AOOs, and that the effects of postulated reactivity accidents will not cause significant damage to the RCPB or impair the capability to cool the core.

Section 4.3 of NUREG-0800 outlines relevant requirements of Commission regulations for this area of review and the associated acceptance criteria, which include the following:

- GDC 13 requires a control and monitoring system to monitor variables and systems over their anticipated ranges for normal operation, AOOs, and accident conditions.

- GDC 26 requires, in part, a reactivity control system capable of holding the reactor subcritical under cold conditions.

- GDC 28 requires, in part, that the reactivity control systems be designed to limit reactivity accidents so that the RCS boundary is not damaged beyond limited local yielding.

In Section 4.3.2.2.6, the applicant modified the text to state, "Online monitoring system is not a required element for a short term reactor operation," and added the statement, "Limits are placed on the axial flux difference so that the heat flux hot channel factor (FQ) is maintained within acceptable limits." In Section 4.3.2.2.9, the applicant modified the text to state, "the in-core and ex-core detector systems provide adequate monitoring of power distributions when the online monitoring system is out of service."

During the review, the staff requested additional information on why online monitoring system outage is not a required element for a short-term operation. In response to RAI-SRP4.3-SRSB-01, the applicant stated the following:

> in the unlikely event that the Online Power Distribution Monitoring System (OPDMS) should become inoperable, reactor operation can continue with shutdown margin and power distribution controls established by bounding analyses and implemented by Technical Specification Limiting Conditions for Operations (LCOs) 3.1.5, 3.1.6, 3.2.1, 3.2.2, 3.2.3 and 3.2.4. These LCOs become applicable immediately or as otherwise specified in the associated Technical Specification applicability statements, when the OPDMS is inoperable.

The staff agrees with the RAI response and finds the proposed change acceptable. In Section 4.3.2.2.6 of the DCD, the phrase "short-term" refers to the time needed to restore the online power distribution monitoring system to operable status.

In the event that the online monitoring system is out of service for a short time, the in-core and ex-core detectors provide the operator with the necessary information regarding the power distribution based on the bounding and precalculated analysis. Therefore, the online monitoring

system is not a required element. The staff finds the proposed change acceptable. The applicant provided a clarification change to Sections 4.3.2.2.4, 4.3.2.2.6, and 4.3.2.2.9 regarding the axial, limiting power distribution, and monitoring instrumentation. The clarification states that, in the event the online monitoring system is out of service, limits placed on axial flux differences are designed so that the heat flux hot channel factor and departure from nuclear boiling ratio (DNBR) are maintained within acceptable limits. The staff finds this clarification to be acceptable.

In Section 4.3.2.4.13, the applicant proposed to change the number of GRCA Ag-In-Cd rodlets from 4 to 12 reduced-diameter rodlets and to change the stainless steel (SS) rodlets from 20 to 12. The total number of rodlets is unchanged at 24. The applicant made a related change to the description of the GRCAs in DCD Section 4.3.2.2.2, Figure 4.3-8 and Figure 4.3-11. Specifically, the applicant changed the gray bank from M0 to MA+MB. This change also applies to the corrections made to the titles of Figures 4.3-8 and 4.3-11. In Section 4.3.2.4.16, the applicant added the statement, "Gray rod operation is a Condition I event which includes the periodic exchange of gray rod banks."

The term "gray rod" refers to the reduced reactivity worth relative to that of an RCCA. The GRCAs are used in load follow maneuvering to provide a mechanical shim reactivity mechanism to eliminate the need for changes to the concentration of soluble boron. The change in the GRCA provides a more distributed absorber material within the assembly. By reducing the diameter of the absorber material, and dispersing the absorber over more rodlets, the reactivity worth of the GRCAs is maintained while lessening the local power perturbation.

This change affects information in Sections 4.3 and 4.3.2.4.13, as well as Table 4.3.1. This change is not expected to impact the conclusion of Chapter 15 and is, therefore, acceptable. Condition I comprises normal operation and operational transients that are accommodated with margin between any plant parameter and the value of that parameter requiring either automatic or manual protective action. The staff agrees with categorizing GRCA operation as a Condition I event and finds the proposed addition of the statement, "gray rod operation is a Condition I event which includes the periodic exchange of gray banks," to Section 4.3.2.4.16 acceptable.

In Section 4.3.2.6.1, "Criticality Design Method Outside the Reactor," the applicant deleted the formula describing the total uncertainty of the criticality calculation, and added the soluble boron credit methodology to Section 4.3.2.6.2. The methodology used in soluble boron credit analysis references WCAP-14416-P (Reference 63). This WCAP is not approved by the NRC; therefore, this proposed change is not acceptable and requires further review by the NRC staff. The staff identified the issue in RAI-SRP4.3-SRSB-03. Since this issue was also identified in RAI-SRP9.1.1-SRSB-05 pertaining to the spent fuel storage rack criticality analysis, the staff designated this as Open Item OI-SRP9.1.1-SRSB-01. In response to RAI-SRP9.1.1-SRSB-05, which also serves as the response to RAI-SRP4.3-SRSB-03, the applicant indicated that it has completely revised the AP1000 spent fuel pool (SFP) criticality analysis in APP-GW-GLR-029, Revision 0, "Spent Fuel Storage Racks Criticality Analysis," with a new methodology that meets 10 CFR 50.68 requirements. The new analysis is APP-GW-GLR-029, Revision 1, "AP1000 Spent Fuel Racks Criticality Analysis" which is a complete rewrite to supersede Revision 0. In addition the applicant has stated that the new method and models replace specific shortcomings that the NRC identified as being no longer reliable as "approved methodology." In a letter dated September 29, 2009, the applicant submitted APP-GW-GLR-029, Revision 2, which is identical to Revision 1, except for minor updates to include pyrex insert burnable absorber design among other burnable absorbers. These new efforts include evaluation of soluble boron. The applicant

also proposed to revise DCD Sections 4.3.2.6.1 and 4.3.2.6.2 to be consistent with the methodology described in APP-GW-GLR-029, Revision 2. The staff evaluation of spent fuel criticality analysis of APP-GW-GLR-029, Revision 2, is addressed in Section 9.1.2.2.4 of this report. The staff confirmed that DCD Sections 4.3.2.6.1 and 4.3.2.6.2 were revised in a subsequent revision to the DCD.

In Section 4.3.2.4.2, the applicant proposed to delete the specific value of the Δ2.22 °Celsius (C) (Δ4 °Fahrenheit (F)) moderator temperature increase that accounts for the control system deadband. The shutdown margin control requirements calculation accounts for the control system deadband and measurement uncertainties by assuming that the moderator temperature is at its maximum possible value before plant trip. This conservatively increases the change in moderator temperature when going from hot full power to hot zero power after plant trip, thereby increasing the shutdown margin control requirement. The uncertainty is based on conservative engineering judgment and includes both instrument errors and deadband. The AP1000 analyses used a preliminary T_{avg} uncertainty, which is provided in the DCD. The applicant revised Section 4.3.2.4.2 to reflect that the allowance for deadband and measurement errors is not set at a fixed value of Δ2.22 °C (Δ4 °F). This change does not affect the conclusion of Chapter 15; therefore, it is acceptable.

The applicant revised Tables 4.3-1 and 4.3-2 and Sections 4.3.2.2.7, 4.3.4, and 4.3.5 to reflect the DCD Chapter 15 accident analysis input assumptions and its results. The applicant revised the fuel assemblies' diameter of guide thimbles and lower part to be consistent with the other dimension values given in the table. The applicant revised the nuclear design parameters, reactivity coefficient, and doppler coefficients in Table 4.3-2 to correctly refer to Figure 15.0.4-1 and to be consistent with the DCD Chapter 15 accident analysis. The applicant revised Section 4.3.4 to reflect COL information updates. The applicant also updated Sections 4.3.2.2.7 and 4.3.5 to correct, delete, and add new references. These changes provide consistency with other DCD sections and do not impact the conclusion of Section 4.3 or Chapter 15; therefore, they are acceptable.

In Table 4.3-2, the applicant provided first-cycle values for the typical hot channel factors and bank worth from beginning of life to end of life. The changes to these values represent first-cycle updated results of the in-core fuel management scheme and are acceptable.

In Section 4.3.2.4.16, the applicant revised mechanical shim load follow and base load operations (including the gray rod bank insertion sequence exchanges) to establish a more negative value than the axial offset associated with all-rods-out condition. The staff considers this to be a conservative change and it is, therefore, acceptable.

The applicant revised Section 4.3.4 to state that APP-GW-GLR-059 (Reference 64) completely addresses the COL information requested in this section, and the DCD incorporates applicable changes. No additional work is required by the COL applicant to address the COL information requested in this section. This change is acceptable as discussed in Section 4.2 of this report.

Revision 17 of DCD Section 4.3 includes numerous editorial and clarification changes concerning specifications that are covered in the core operating limits report, general formatting changes to the references, and clarifications to the variable units used in Section 4.3 of the DCD. The staff finds these clarifications to be acceptable.

4.3.2 Conclusion

The staff reviewed the changes to DCD Revision 17 that demonstrate that sufficient control rod and burnable poison worth exist to provide safe shutdown of the plant. Furthermore, the control rod system is designed to ensure that reactivity accidents do not result in damage to the RCPB. Therefore, the staff finds that the proposed DCD changes comply with the requirements of GDC 13, 26, and 28 and are acceptable. The staff expects the above changes to apply to all COL applications referencing the AP1000 DC.

4.4 Thermal-Hydraulic Design

In Revision 17 to the AP1000 DCD, the applicant proposed changes related to the reactor core thermal-hydraulic design. In a letter dated October 31, 2006, the applicant submitted TR-18, which provides rationale and justification for the proposed changes related to the reactor core and fuel design. In TR-108, APP-GW-GLN-019, "AP1000 Standard Combined License Technical Report, Fluid System Changes," the applicant proposed revisions to the AP1000 fluid systems, including the digital metal impact monitoring system (DMIMS). DCD Revision 17, Section 4.4, includes the proposed changes that affect the reactor core thermal-hydraulic design.

In its letter of May 26, 2007, regarding its application to amend the AP1000 DC rule, the applicant referred to the criterion of 10 CFR 52.63(a)(1)(vii) and stated that these proposed changes contribute to increased standardization of the certification information.

4.4.1 Evaluation

DCD Section 4.4, "Thermal and Hydraulic Design," describes the AP1000 reactor core thermal-hydraulic design to ensure adequate heat removal to prevent fuel damage during normal operation and transients. In Revision 17 of the AP1000 DCD, the applicant proposed the following changes to Section 4.4 related to the AP1000 core thermal-hydraulic design: (1) revision of the core bypass flow; (2) change of the maximum rod bow penalty for DNBR calculation from less than 1.5 percent DNBR to less than about 2 percent DNBR; (3) change of the hydraulic loads calculation from the mechanical design flow to the best estimate flow; (4) change of the peak linear power resulting from overpower transient/operator errors from 73.82 kW/m (22.5 kW/ft) to less than or equal to 73.65 kW/m (22.45 kW/ft); (5) change of the term "canned motor pump" to "reactor coolant pump"; (6) revision of the description of the DMIMS; (7) changes of the values of several parameters in Table 4.4-1; (8) addition of WCAP-15063-P-A, Revision 1; (9) addition of Reference 87 (WCAP-16652-NP); and (10) division of Section 4.4.7, "Combined License Information," into Sections 4.4.7.1 and 4.4.7.2, with revisions to each section.

GDC 10 specifies that the reactor core and the associated coolant, control, and protection systems must be designed with appropriate margin to ensure that SAFDLs are not exceeded during any condition of normal operation, including the effects of AOOs. In its review of the AP1000 DCD changes related to DCD Section 4.4, the staff used the guidance of NUREG-0800 Section 4.4, "Thermal and Hydraulic Design," which sets forth the acceptance criteria used by the staff to evaluate the thermal-hydraulic design of the reactor core for compliance with the relevant requirements of GDC 10. The following discusses the evaluation of these changes.

DCD Section 4.4.1.3.1, "Design Basis," identifies the core bypass flow (which is not considered effective for heat removal) as the coolant flow through the thimble tubes and the leakage from

the core barrel shroud region into the core. Revision 17 revised the core bypass flow as the coolant flow through the thimble and instrumentation tubes and the leakage between the core barrel and core shroud, head cooling flow, and leakage to the vessel outlet nozzles. In TR-18, the applicant indicated that it made this change to accurately describe the core cavity flow area caused by a change from the core baffle-former design to a welded core shroud design, which eliminates the need for bolts in the high fluence regions immediately adjacent to the reactor core. The revision to DCD Section 4.4.1.3.1 clarifies the regions of the core bypass flow that are not effective for core heat transfer. The shroud core cavity flow is considered to be active flow that is effective for fuel rod cooling. Therefore, the applicant deleted Item E in Section 4.4.4.2.1 from a core bypass flow path to be consistent with core bypass flow design basis detailed in Section 4.4.1.3.1. This revision results in slight changes in the effective core flow area, average velocity along the fuel rods, and average mass velocity, as reported in Tables 4.1-1 and 4.4-1, from 3.85 square meters (m^2) (41.5 square feet (ft^2)), 4.85 meters per second (m/s) (15.9 feet per second (ft/s)), and 3268.5 kilograms per second-square meter (kg/s-m^2) ($2.41x10^6$ pounds per hour-square feet) (lbm/hr-ft^2)) to 3.88 m^2 (41.8 ft^2), 4.82 m/s (15.8 ft/s), and 3458.4 kg/s-m^2 ($2.55x10^6$ lbm/hr-ft^2), respectively.

DCD Section 4.4.4.2.1 states that calculations using drawing tolerances in the most conservative direction and accounting for uncertainties in the pressure losses show the core bypass to be no greater than the 5.9-percent design value. The maximum value of 5.9 percent allotted as bypass flow and the thermal design flow of 94.1 percent of the thermal flow rate assumed for the core cooling evaluations, as stated in DCD Sections 4.4.1.3.2 and 4.4.1.3.1, respectively, remain unchanged. Thus, the change in the core bypass flowpaths has insignificant effects on the core cooling calculation and is, therefore, acceptable. The change in the average mass velocity reflects a correction to the coolant temperature used in the calculation of the core average flow rate from the core average temperature to the core inlet temperature. This is acceptable because the calculation with higher water density of the core inlet temperature results in a higher mass velocity with the same volumetric flow rate.

In DCD Revision 17, the applicant added Reference 88 to Sections 4.4.1.2.1 and 4.4.8 (the reference section). Reference 88 addresses the NRC approval of the maximum burnup limit of 62,000 MWD/MTU for WCAP-10444-P-A, WCAP-12610-P-A, WCAP-12488-A, and WCAP-15063-P-A, all of which are referenced in Section 4.4.8. The applicant added Reference 88 for completeness, and the staff finds this addition acceptable.

In DCD Revision 17, the applicant added Reference 82a to Sections 4.4.2.2.1 and 4.4.8. Reference 82a addresses the application of an adjustment factor to the WRB-2M critical heat flux correlation described in Reference 82, which is used in the AP1000 thermal-hydraulic design calculations. The applicant added Reference 82a for completeness, and the staff finds this addition acceptable.

Section 4.4.2.2.5 states that the maximum rod bow penalties accounted for in the design safety analysis are based on an assembly average burnup of 24,000 MWD/MTU and indicated that the maximum rod bow penalty is less than 1.5 percent DNBR. In Revision 17, the applicant changed the maximum rod bow penalty of "less than 1.5 percent DNBR" to "less than about 2 percent DNBR." In TR-18, the applicant stated that it made this change for consistency with the basis used for 3-D flow-accelerated corrosion (FAC) analysis. Because the change of the maximum penalty from 1.5 percent to about 2 percent DNBR is a more conservative change, and the actual value of the rod bow penalty used in the safety analysis will be calculated based on actual fuel design and assembly burnup of 24,000 MWD/MTU, the staff concludes that the change is acceptable.

Section 4.4.2.6.2 states that the hydraulic loads at normal operating conditions are calculated considering the mechanical design flow and account for the minimum core bypass flow based on manufacturing tolerances. In Revision 17, the applicant changed the mechanical design flow to the best estimate flow. As stated in TR-18, the applicant changed the mechanical design flow to the best estimate flow for the hydraulic calculation to achieve consistency with the current design procedures. Therefore, the staff finds the change to be acceptable.

In DCD Revision 17, Section 4.4.2.7.1, the applicant corrected a typographic error in the Dittus-Boelter correlation for the force convection heat transfer coefficients calculation. However, the staff found that the correction still includes editorial errors. The applicant stated, in response to a telecom of November 17, 2008, that it will correct these errors in a follow up version of DCD Section 4.4.2.7.1. In a subsequent revision to the DCD, the applicant made appropriate changes to DCD Section 4.4.2.7.1, which resolves this issue.

In Section 4.4.2.11.6, the applicant changed the peak linear power resulting from overpower transient or operator errors from 73.82 kW/m (22.5 kW/ft) to less than or equal to 73.65 kW/, (22.45 kW/ft). This change ensures consistency with the reactor design value specified in Tables 4.1-1 and 4.4-1 and is, therefore, acceptable.

In Section 4.4.4.6, the applicant changed the term "canned motor pump" to the more generic term, "reactor coolant pump." This editorial change does not affect the negative slope for the pump head-capacity curve as a generic reactor coolant pump characteristic and is, therefore, acceptable.

In Section 4.4.6.4, the applicant made changes to the description of the DMIMS. In TR-103, APP-GW-GLN-019, Revision 2, "AP1000 Standard Combined License Technical Report, Fluid Systems Changes," the applicant stated that it made these changes to correct a misinterpretation of the required number of sensors and to correct and delete incorrect information. The requirement for loose parts monitoring system sensors was incorrectly interpreted as requiring four, rather than two, sensors per collection region. This correction results in the removal of the term "redundancy" regarding sensors at each RCS location and instrumentation channel. The applicant made other changes to the descriptions of the DMIMS performance tests method and the technique used to minimize false impact detection. The applicant made these changes to accurately represent the DMIMS design, and they do not result in a change of the design. The DMIMS design continues to conform to system design aspects of Regulatory Guide (RG) 1.133, Revision 1, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors."

In Table 4.4-1, the applicant changed the values of several parameters to be consistent with Table 4.1-1. In Footnote j, the applicant revised the theoretical density of the fuel from 95 percent to 95.5 percent for AP1000 (95 percent for others). The applicant modified Footnote f, which is associated with heat transfer, by changing the phrase, "based on 157 fuel assemblies and hot densified fuel length" to "based on densified active fuel length." The applicant rounded the value for AP1000 to 18.77 kW/m (5.72 kW/ft). The changes regarding the theoretical density are acceptable as they are consistent with DCD Section 4.2.3.2.

In Section 4.4.2.11, the applicant added Reference 85, WCAP-15063-P-A, Revision 1, "Westinghouse Improved Performance Analysis and Design Model (PAD 4.0)." This report presents an NRC-approved fuel rod design methodology. The addition of a reference to this

topical report reflects the NRC's approval of the methodology. Therefore, this addition is acceptable.

DCD Revision 16, Section 4.4, proposed to change Reference 9, WCAP-12488-A, from Tier 2* information to Tier 2. As discussed in Section 4.1 of this report, in response to RAI-SRP4.2-SRSB-01, the applicant stated that it would withdraw the request that this Tier 2* item be reassigned to Tier 2 status in Chapter 4. In DCD Revision 17, the applicant restored all Tier 2* information identified in DCD Revision 15, including WCAP-12488-A, to Tier 2*. The staff finds this acceptable.

In Revision 17, the applicant revised Section 4.4.7.1 to state that APP-GW-GLR-059 (Reference 87) completely addressed the COL information requested in this section and the DCD incorporated applicable changes; therefore, no additional work is required by the COL applicant to address the COL information requested in this section. The staff finds this revision to be acceptable, as discussed in Section 4.2 of this report.

In DCD Revision 17, Section 4.4.7.2 states the following:

> Following selection of the actual plant operating instrumentation and calculation of the instrumentation uncertainties of the operating plant parameters as discussed in Subsection 7.1.6, and prior to fuel load, the Combined License holder will calculate the design limit DNBR values. The calculations will be completed using the RTDP with these instrumentation uncertainties and confirm that either the design limit DNBR values as described in Section 4.4, "Thermal and Hydraulic Design," remain valid, or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties, such as rod bow penalty.

This is a change from DCD Revision 15, Section 4.4.7, "Combined License Information." The applicant has changed the "combined license applicant" who will calculate the design limit DNBR values to the "combined license holder" who will calculate the design limit DNBR values before fuel load. In response to RAI-SRP4.4-SRSB-02, the applicant stated that the design limit DNBR depends on the selection of the actual plant operating instrumentation, as well as on calculation of the instrumentation uncertainties of the operating plant parameters. The applicant further stated the following:

> The actual calculated instrumentation uncertainties will be used, when available, to recalculate the RTDP [revised thermal design procedures] DNBR design limits. If these recalculated DNBR design limits are less than those given in DCD 4.4.1.1.2, then the DCD values are conservatively high and will not be revised. If the calculated values are higher than the values in the DCD, then the design limits will be revised. Note that since conservatively high values of the instrumentation uncertainties were used in the calculations, we do not expect the design limits to change.

The staff agrees that the actual design limit DNBR can only be calculated after the selection of actual plant operating instrumentation and, therefore, the change from "the combined license applicant" to the "combined license holder" who will calculate actual design limit DNBR values is acceptable. It should be noted that in NUREG-1793, Appendix F, "Combined License Action Items," Item 4.4-1 reiterated DCD COL Information Item 4.4-2. Therefore, this same change in COL Information Item 4.4-2, as described in DCD Section 4.4.7.2, is applied to NUREG-1793

Item 4.4-1. Also, in Revision 17 of the DCD, the applicant revised Table 1.8-2, "Summary of AP1000 Standard Plant Combined License Information Items," to be consistent with the changes to Section 4.4.7.2 to indicate that COL Information Item 4.4-2 is an action required by the COL holder. This is acceptable to the staff.

4.4.2 Conclusion

The staff has reviewed the changes to DCD Section 4.4. Based on the evaluation discussed above, the staff concludes that the core thermal-hydraulic design continues to meet the requirements of GDC 10 and is, therefore, acceptable.

4.5 Reactor Materials

4.5.1 Control Rod Drive System Structural Materials

4.5.1.2.1 Material Specification

In Revision 16 to the AP1000 DCD, the applicant proposed changes to Section 4.5.1 to include austenitic SS Types 304, 304L, 316, and 316L for parts of the control rod drive mechanisms (CRDMs) and control rod drive (CRD) line exposed to reactor coolant, including pressure-boundary components. TR-33, APP-GW-GL-009, "Pressure Boundary Material Change," Revision 1, submitted in a letter dated May 24, 2007, identified and justified these changes.

In addition, the applicant revised DCD Tier 2, Section 4.5.1, to include the use of a cobalt alloy or qualified substitute to fabricate CRDM latches and links. The applicant also modified the nickel-chromium-iron alloy (Alloy 750) specification used for CRDM springs from Aerospace Material Specification (AMS) 5698E and AMS 5699E to AMS 5698 and AMS 5699. TR-106, APP-GW-GLN-106, "AP1000 Licensing Design Changes for Mechanical Systems and Component Design Updates," Revision 1, submitted in a letter dated September 28, 2007, identified and justified the aforementioned changes.

4.5.1.2.2 Evaluation

4.5.1.2.2.2 Changes to Control Rod Drive Mechanism Stainless Steel Materials (TR-33)

GDC 1, "Quality Standards and Records," requires structures, systems, and components (SSCs) important to safety to be designed, fabricated, erected, and tested to quality standards to ensure a quality product in keeping with the required safety function. GDC 4, "Environmental and Missile Dynamic Effects Design Bases," requires that SSCs important to safety be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure. The staff review of the AP1000 DCD changes related to this section ensures that the materials for the CRD system meet the American Society of Mechanical Engineers (ASME) Code requirements and are compatible with the reactor coolant environment to ensure a quality product commensurate with its importance to safety.

DCD Tier 2, Section 4.5.1, describes the materials used to fabricate components of the CRDM and CRD line. The parts of the CRDMs and CRD line exposed to reactor coolant are made of materials designed to resist the degradation mechanisms of the reactor environment.

DCD Tier 2, Section 4.5.1, describes the materials used to fabricate components of the CRDM. The parts of the CRDM exposed to reactor coolant are made of materials designed to resist the degradation mechanisms of the reactor environment. Currently, DCD Section 4.5.1.1 and the corresponding Table 5.2-1 in Section 5.2.3 include SS Types 304LN and 316LN, which have high resistance to sensitization. Therefore, these materials are more resistant to stress-corrosion cracking (SCC) because of their low carbon content. The applicant revised DCD Tier 2, Section 4.5.1, to include austenitic SS Types 304, 304L, 316, and 316L, as discussed in TR-33. TR-33, which provides the basis for the change, states that the addition of these materials will enhance manufacturing flexibility, reduce costs, and reduce risk relative to material availability. The applicant submitted changes to the AP1000 DCD, as proposed in TR-33, pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that the proposed changes contribute to increased standardization of the certification information.

SS Types 304 and 316 (higher carbon content) are less resistant to sensitization due to heat treatment or welding. In addition, NRC Information Notice 2006-27, "Circumferential Cracking in Stainless Steel Pressurizer Heater Sleeves of Pressurized Water Reactors," and numerous requests for relief from the ASME Code concerning repairs to leaking CRDM canopy seal welds discuss emerging issues involving SS Types 304 and 316. These instances of SCC are occurring in stagnant or dead end pressurized-water reactor (PWR) coolant environments. Since Type 304 and 316 SS materials are more susceptible to intergranular stress-corrosion cracking (IGSCC) and transgranular stress-corrosion cracking than the low-carbon SS Types 304L, 304LN, 316L, and 316LN, the use of Types 304 and 316 materials may affect the integrity of the CRD components (including the RCPB portions of the latch housing and rod travel housing). Specifically, the use of these materials can affect the structural integrity of CRD components that are subjected to stagnant water (trapped oxygen), dead legs, or areas prone to increased levels of oxygen.

Therefore, the staff requested, in RAI–SRP4.5.1-CIB1-01, that the applicant delete the proposed addition of SS Types 304 and 316 from the AP1000 DCD or provide further justification addressing the acceptability of the proposed addition of these materials.

In a letter dated May 30, 2008, the applicant provided further justification concerning the acceptability of the proposed addition of SS Types 304 and 316 in components subjected to stagnant water. The applicant stated that all austenitic SSs are procured in the solution annealed condition and that controls are established in Section 5.2.3.4 of the AP1000 DCD to avoid sensitization due to heat treatment or welding and prevent susceptibility to intergranular attack as addressed by RG 1.44, "Control of the Use of Sensitized Stainless Steel," issued May 1973. In addition, cleaning procedures and contamination preventative measures are implemented to prevent the presence of detrimental impurities that could contribute to SCC. The AP1000 CRDM design uses only one canopy seal (CRDM to the rod travel housing) in lieu of the three canopy seals in current designs. In addition, this canopy seal weld was redesigned with a larger radius and thicker wall to reduce stress levels, and the design includes a vent and drainpath for the canopy seal volume to prevent fully stagnant conditions. The number of occurrences of cracking in SS components exposed to PWR operating environment is small considering the number of SS components used in PWR applications. Most of these failures have occurred as a result of adverse conditions (trapped oxygen in stagnant water) and high stresses.

The staff agrees that using the guidance in RG 1.44 and proper cleaning techniques limits the amount of sensitization of Types 304 and 316 SS, which in turn reduces the susceptibility to SCC. In addition, the redesign of the CRDM reduces the stresses and adverse environment

(vent and drainlines to minimize the presence of an oxygenated environment) that is a major contributor to the susceptibility of the SS to SCC. Therefore, the staff finds that Types 304 and 316 SS can be used, in addition to Types 304L, 304LN, 316L, and 316LN (which are less susceptible to sensitization) previously approved by the staff in NUREG-1793 because the applicant's design changes will reduce the number of canopy seal welds to one per CRDM, reduce the stresses in the canopy seal weld, eliminate the presence of an oxygenated environment using vent and drainlines, and follow the guidance in RG 1.44. However, these design changes (reduced number of welds, reduced stresses, and use of vent and drainlines) and the use of RG 1.44 are critical in preventing the occurrence of SCC in these components. However, the staff concluded that the applicant should add these design changes and the use of RG 1.44 for the CRDM components, including the canopy seal welds, to the DCD. It should be noted that DCD Section 4.5.1.2 applies the controls on preventing SCC, including the guidance in RG 1.44 for the austenitic SS pressure-housing components of the CRDM. The canopy seal welds may not be considered a pressure-housing component since they only provide a leakage barrier. The staff identified this issue as Open Item OI-SRP4.5.1-CIB1-01. In a letter dated May 13, 2009, the applicant provided a marked-up copy of the DCD, which included the design changes and the use of RG 1.44 as requested by the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The identified added materials and the additional information on grades, types, or classes meet the requirements of GDC 1, GDC 4, and the ASME Code, Sections II and III, in accordance with the guidance of NUREG-0800 Section 4.5.1. Therefore, the addition of the identified materials and the additional information on grades, types, or classes presented in TR-33, Revision 1, is acceptable.

The staff reviewed the proposed material changes to the AP1000 DCD. The applicant has incorporated the proposed changes, as identified in TR-33, Revision 1, into the AP1000 DCD. Therefore, the staff finds that the DCD changes that address material changes, as proposed by the applicant in TR-33, Revision 1, are acceptable.

The proposed changes contribute to the increased standardization of the certification information in the AP1000 DCD and are acceptable.

4.5.1.2.2.3 Changes to Control Rod Drive Mechanism Latches, Links, and Springs (TR-106)

GDC 26 requires that one of the reactivity control systems use control rods, preferably with a positive means for inserting the rods, and be capable of reliably controlling reactivity changes for assurance that fuel design limits are not exceeded under conditions of normal operation, including AOOs.

The staff reviewed the AP1000 DCD changes related to this section to ensure that the materials for the CRDMs will perform adequately throughout the design life of the plant. The CRDM latches, links, and springs are not RCPB components and are, therefore, not required to be fabricated from materials meeting ASME Code, Section III. DCD Tier 2, Section 4.5.1, describes the materials used to fabricate CRDM components.

The applicant revised DCD Tier 2, Section 4.5.1, to include a cobalt alloy or qualified substitute to fabricate CRDM latches and links. Previously, the applicant intended to fabricate latches and links from SS hard faced with Stellite.™ The applicant also deleted its requirement that all cobalt alloy material be ordered in the solution-treated, cold-worked condition. In addition, the

applicant modified the nickel-chromium-iron alloy (Alloy 750) specification used for CRDM springs. The applicant also changed the specification from AMS 5698E and AMS 5699E to AMS 5698 and AMS 5699.

In TR-106, the applicant stated that the previous design of the CRDM in Section 4.5 of Revision 15 of the AP1000 DCD was based on a design utilized in past Westinghouse plants to achieve a design life of 3 million steps. The applicant modified the design to meet a design life of 8 million steps, which required a material change for CRDM latches and links. The redesigned latches have also been modified to a double tooth design constructed of solid Stellite™ in lieu of the previous design, which utilized a single tooth design and was constructed of SS hard faced with Stellite. ™

In response to RAI-TR106-ClB1-02, dated October 5, 2007, the applicant provided additional information regarding the use of solid Stellite™ components in PWR CRDMs. The applicant stated that cast single-tooth latches constructed from solid Stellite™ have been used extensively in the Combustion Engineering (CE) fleet and all CE units in Korea. CE has used single-tooth cast Stellite™ to construct latches for the past 30 years in its units. Electricite de France (EDF) has used double-tooth hardfaced latches. The AP1000 cast latches will be based on AMS 5387.

AMS 5387 (similar to Stellite™ 6) is a cast cobalt-based alloy that provides adequate resistance to wear and corrosion in the reactor coolant environment. The staff is unaware of any failures or degradation issues associated with the use of cast Stellite™ in CRDM components. Based on the corrosion resistance and mechanical properties of this material, as well as the favorable operating experience in currently operating plants, the staff finds the applicant's use of AMS 5387 to fabricate latches and links acceptable. The applicant's elimination of required solution heat treatment and cold-working for cobalt alloys used in the CRDMs is acceptable because the AMS 5387 used for latches and links requires that components fabricated to this specification are used in the as-cast condition. It is the staff's understanding that cobalt alloy link pins will still be delivered in the solution treated, cold-worked condition because these components are not cast.

In addition to the modifications to DCD Tier 2, Section 4.5.1.3, discussed above, the applicant changed the specification for CRDM springs (Alloy 750) from AMS 5698E or AMS 5699E to AMS 5698 or AMS 5699. This modification does not change the materials that the staff previously approved. The modification to specifications listed for CRDM springs allows current versions of AMS 5698 and AMS 5699 to be used and is, therefore, acceptable.

The staff finds that the proposed modifications are acceptable and meet the requirements of GDC 26 and the acceptance criteria of NUREG-0800 Section 4.5.1. In addition, the NRC staff reviewed the proposed changes as they relate to Revision 16 of the AP1000 DCD. Revision 16 of the DCD incorporates the proposed changes identified in TR-106. Accordingly, these changes are generic and are expected to apply to all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plants. Thus, the proposed changes contribute to the increased standardization of the AP1000 certified design and, therefore, meet the requirements of 10 CFR 52.63(a)(1)(vii).

4.5.1.2.3 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to DCD Section 4.5.1 meet the requirements of 10 CFR 50.55a, "Codes and standards," and

ASME Code, Section III, and are, therefore, acceptable. The proposed changes have been incorporated into the AP1000 DCD. Furthermore, the staff finds that the TR-33, Revision 1, conclusions regarding the evaluation of the changes to the material specification are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information. In addition, based on the above evaluation, the staff finds that the revisions proposed by the applicant to AP1000 DCD, Section 4.5.1, meet the requirements of GDC 26 and, therefore, are acceptable. The AP1000 DCD incorporates the proposed changes, as identified in TR-106, Revision 1. Furthermore, the staff finds that the TR-106, Revision 0, conclusions regarding the evaluation of the changes to the material specification are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

4.5.2 Reactor Internal and Core Support Materials

In Revision 16 of the AP1000 DCD, the applicant proposed design changes related to the material specifications for AP1000 reactor internal components. This resulted in changes to Section 4.5.2.1. In a letter dated July 31, 2007, the applicant submitted TR-31, APP-GW-GLN-015, "Reactor Internals Material Changes for the AP1000 Plant," Revision 1, which included WCAP-16624-P, Revision 1, "Reactor Internals Materials Changes for the AP1000 Plant," to provide the technical justification for the proposed design changes related to the material specifications for reactor internal components.

In TR-31, the applicant proposed the following changes to the AP1000 reactor internal and core support material specifications: (1) the addition of American Iron and Steel Institute Types 304, 304H, and 304L SSs to the material specifications for reactor internal core structure components; (2) the use of nickel-based Alloys 718 and 750 for the clevis insert-to-vessel bolts in place of strain-hardened Type 316 SS; (3) the use of nickel-based Alloy 690 for the clevis inserts in place of 304-series SSs; (4) the addition of Stellite™ 6 to the hardfacing materials for the radial keys, clevis inserts, and alignment pins; and (5) the use of nickel-based Alloy 750 for the irradiation specimen springs in place of Type 302 SS.

Other related changes identified in TR-31 include: (1) an additional specification in DCD Section 4.5.2.1 for qualification of welding procedures in accordance with the staff guidance in RG 1.44; (2) revisions to the language in DCD Section 4.5.2.1 pertaining to the susceptibility of reactor internal components to irradiation-assisted stress-corrosion cracking (IASCC) and void swelling; and (3) an additional statement in DCD Section 4.5.2.1 indicating that Alloy 600 would not be used in the AP1000 reactor internal components.

Revision 16 to DCD Section 4.5.2.1 implements all of the above changes proposed in TR-31 as follows:

> The major core support structure material for the reactor internals is SA-182, SA-479, or SA-240 Types 304, 304L, 304LN, or 304H stainless steels. Fabricators performing welding of any of these materials are required to qualify the welding procedures for maximum carbon content and heat input for each welding process in accordance with RG 1.44. For threaded structural fasteners the material used is strain hardened Type 316 stainless steel and for the clevis insert-to-vessel bolts either Unified Numbering System for Metals and

Alloy UNS N07718 or N07750. Remaining internals parts not fabricated from Types 304, 304L, 304LN, or 304H stainless steels typically include wear surfaces such as hardfacing on the radial keys, clevis inserts, alignment pins (Stellite™ 6 or 156 or low cobalt hardfaces); dowel pins (Type 316); hold down spring (Type 403 stainless steel (modified)); clevis inserts (UNS N06690); and irradiation specimen springs (UNS N07750). Core support structure and threaded structural fastener materials are specified in the ASME Code, Section III, Appendix I as supplemented by Code Cases N-60 and N-4. The qualification of cobalt free wear resistant alloys for use in reactor coolant is addressed in Subsection 4.5.1.3.

The use of cast austenitic stainless steel (CASS) is minimized in the AP1000 reactor internals. If used, CASS will be limited in carbon (low carbon grade: L grade) and ferrite contents and will be evaluated in terms of thermal aging effects.

The estimated peak neutron fluence for the AP1000 reactor internals has been considered in the design. Susceptibility to irradiation-assisted stress corrosion cracking or void swelling in reactor internals identified in the current pressurized water reactor fleet are being addressed in reactor internals material reliability programs. The selection of materials for the AP1000 reactor internals considers information developed by these programs. Ni-Cr-Fe Alloy 600 is not used in the AP1000 reactor internals.

4.5.2.1 Evaluation

GDC 1 and 10 CFR 50.55a require that SSCs important to safety be designed, fabricated, and tested to quality standards commensurate with the importance of the safety function to be performed. The staff's review of the AP1000 DCD changes related to this section covers material, components design, fabrication, and inspection to ensure structural integrity in compliance with 10 CFR 50.55a and GDC 1.

TR-31 proposes changes to the material specifications for the AP1000 reactor internal components. DCD Tier 2, Section 4.5.2.1, identifies the reactor internal material specifications. In Revision 15 of the DCD, the major materials of construction for the reactor internal core support structure components, excluding threaded structural fasteners, are Type 304LN SS. Revision 15 of the DCD specifies strain-hardened Type 316 SS for threaded structural fasteners. Other reactor internal materials specified in DCD Revision 15, Section 4.5.2.1, include Stellite™ 156 or low cobalt hardfacing on the radial keys, clevis inserts, and alignment pins; Type 316 SS for the dowel pins; modified Type 403 SS for the holddown spring; and Type 302 SS for irradiation specimen springs.

The proposed material specification changes primarily involve the addition of Types 304, 304H, and 304L to the 304-series SSs specified for the reactor internal core support structure components. Other material changes include: (1) the use of nickel-based Alloys 718 and 750 for the clevis insert-to-vessel bolts in place of strain-hardened Type 316 SS; (2) the use of nickel-based Alloy 690 for the clevis inserts in place of 304-series SSs; (3) the addition of Stellite™ 6 to the above hardfacing materials for the radial keys, clevis inserts, and alignment pins; and (4) the use of nickel-based Alloy 750 for the irradiation specimen springs in place of Type 302 SS. DCD Revision 16, Section 4.5.2.1, implements the changes to these material specifications.

Section 3 of TR-31 provides a technical description and justification for the proposed material specification changes. The addition of Types 304 and 304H SSs is significant because these SSs have higher carbon content than Type 304LN. Types 304 and 304H have a maximum carbon content of 0.08 percent (by weight) and 0.10 percent, respectively; whereas, Type 304LN is a low-carbon SS with a maximum carbon content of 0.03 percent. Type 304H, in particular, is notable as a high-carbon SS with a specified minimum carbon content of 0.04 percent. None of the other SS grades specified in DCD Revision 16, Section 4.5.2.1 (Types 304, 304L, 304LN), has a specified minimum carbon content. The 304H grade was developed to ensure better resistance to high-temperature creep by maintaining at least 0.04-percent carbon. The higher allowable carbon content in Types 304 and 304H austenitic materials can potentially result in a significant degree of sensitization to intergranular corrosion and IGSCC at elevated temperatures. In the temperature range of 426.7 °C to 815.6 °C (800 °F to 1,500 °F), chromium carbides, $(Fe,Cr)_{23}C_6$, are insoluble and precipitate at grain boundaries through the diffusion of carbon. Precipitation of chromium carbides at grain boundaries result in the depletion of chromium in the surrounding matrix alloy immediately adjacent to the grain boundaries. The chromium-depleted alloy at the grain boundaries is much less corrosion resistant than the rest of the bulk alloy (i.e., away from the grain boundaries). The galvanic coupling of chromium-depleted alloy at the grain boundaries with bulk alloy in the passive state (due to undepleted chromium) can result in significant intergranular corrosion. High-carbon SSs subjected to temperatures in the range of 426.7 °C to 815.6 °C (800 °F to 1,500 °F) for a sufficient time to allow for the formation of chromium carbides at the grain boundaries are sensitized to intergranular corrosion and IGSCC. When temperatures exceed 815.6 °C (1,500 °F), the chromium carbides are soluble, and below 426.7 °C (800 °F) the diffusion rate of carbon is not sufficient to permit the formation of chromium carbides. Therefore, it is specifically in the intermediate temperature range that sensitization is a significant concern for high-carbon SSs. Welding is known to produce sensitization in weld heat affected zones (HAZs), located on either side and, at times, somewhat removed from the actual weld bead. It is in these HAZs that the welding process can produce temperatures in this intermediate range for a sufficient time to allow for carbon diffusion and the formation of chromium carbides and chromium-depleted zones at the grain boundaries. The carbon content of low-carbon SS grades (i.e., SSs with a specified maximum carbon content of 0.03 percent) is not high enough for sensitization to be a significant concern because a sufficient quantity of carbon does not exist to cause significant chromium depletion at grain boundaries within a practical timeframe for the welding process.

High-carbon austenitic SSs in nuclear reactor structural components that have become locally sensitized are potentially susceptible to IGSCC at these sensitized locations. While significant intergranular corrosion is generally not an issue for unstressed parts, the presence of tensile stresses has been known to produce IGSCC in components where sensitization has occurred. For a given high-carbon SS component, the effects of sensitization can be minimized by controlling weld parameters, such as heat input and cooling rate. RG 1.44 describes acceptable methods for controlling the processing of SSs to avoid sensitization that could lead to IGSCC. The RG specifies that, for a given material composition, welding practices should be controlled to avoid excessive sensitization of base metal HAZs adjacent to welded joints. The RG also specifies an intergranular corrosion test for the qualification of welding procedures to be used for welding SSs having a carbon content of greater than 0.03 percent. DCD Revision 16, Section 4.5.2.1, includes language requiring that welding procedures be qualified in accordance with RG 1.44.

DCD Revision 16, Section 4.5.2.1, also added Type 304L, a low-carbon grade of SS (0.03-percent maximum) similar to Type 304LN, to the list of permissible SS grades for reactor internal core support structure components. Type 304L has a lower nitrogen content (0.10-percent maximum) than Type 304LN (0.10 percent to 0.16 percent). Because of its lower nitrogen content, Type 304L is not as strong as Type 304LN and high-carbon SS Types 304 and 304H. Relative to Type 304L, the higher nitrogen content in Type 304LN results in a strengthened material that is both resistant to sensitization (due to the low carbon content) and possesses the higher tensile and yield strength properties of Type 304 and 304H materials. Types 304LN, 304, and 304H all possess an ASME Code minimum tensile strength of 517.1 megapascals (MPa) (75 kilopounds per square inch (ksi)) and a minimum yield strength of 206.8 MPa (30 ksi). Type 304L possesses an ASME Code minimum tensile strength of 475.7 MPa (69 ksi) and a minimum yield strength of 172.4 MPa (25 ksi). Type 304L SS may be used for reactor internal core support structural applications where its lower strength properties are permitted, in accordance with ASME Code, Section III.

According to the applicant, the primary justification for adding Types 304, 304H, and 304L SSs to DCD Section 4.5.2.1 is the application of these materials in currently operating Westinghouse plants. The applicant indicated that SCC has not been experienced in reactor internal core support structure components fabricated with any of these three materials. The use of Type 304L, the lower strength grade, is unconditionally approved for currently operating Westinghouse reactors where its lower strength properties are permitted. The applicant stated that it previously implemented a change from Type 304 to Type 304H for certain reactor internal components in operating plants as the reactor design evolved over time; the applicant provided a list of plants to demonstrate the extensive application of Type 304H SS in reactor internal core support structure components for these later designs. The applicant noted that, for many of the later plants, carbon content in several Type 304 components was limited to a specified range of 0.04 percent to 0.08 percent. This was stated as being the equivalent of a Type 304H SS with a more restrictive 0.08-percent upper limit on carbon content. The applicant stated that, to be consistent with these later plants, the carbon content in Type 304H SSs should be limited to a maximum of 0.08 percent for reactor internal core support structure components in the AP1000 plant. The Westinghouse Utilities Requirements Document requires the use of RG 1.44. Accordingly, all fabricators will be required to establish maximum heat inputs for each welding process with respect to the maximum carbon content for each SS type.

Overall, the staff found that the applicant provided sound justification for the addition of Types 304, 304H, and 304L SSs to the material specifications in DCD Revision 16, Section 4.5.2.1. In particular, the staff noted that the currently operating Westinghouse plants have not experienced problems with IGSCC in reactor internal core support structure components fabricated from these materials. Furthermore, the staff noted that DCD Revision 16, Section 4.5.2.1, specifies that fabricators performing welding of any of these materials are required to qualify the welding procedures for maximum carbon content and heat input for each welding process in accordance with RG 1.44.

In an RAI dated March 11, 2007, the staff requested that the applicant clarify or elaborate on several issues. In RAI-TR31-001, Question 1, Part a, the staff noted an inconsistency in the language in Section 1.2, "Introduction and Brief Description of Change," of TR-31. Specifically, the staff noted that Section 1.2 of TR-31 states that DCD Revision 15, Section 4.5.2.1, currently specifies Type 304LN SS for reactor internal core support structure components. However, Section 1.2 also states that reactor internal components were designed using Types 304, 304H, and 304L SSs. Therefore, the staff requested that the applicant clarify whether it changed the AP1000 design for the reactor internal components after the issuance of DCD Revision 15 to

include these additional SS grades. In its response to RAI-TR31-001, Question 1, Part a, the applicant indicated that it did change the design of the AP1000 reactor internal components after the issuance of Revision 15 of the DCD to include Types 304, 304H, and 304L SSs in addition to Type 304LN SS as potential materials of construction for AP1000 reactor internal components. The staff concludes that this response resolved RAI-TR31-001, Question 1, Part a, because the applicant adequately clarified the statement made in Section 1.2 of TR-31 pertaining to reactor internal component design.

In RAI-TR31-001, Question 1, Part b, the staff requested that the applicant list the materials of construction for each reactor internal component based on the newly proposed reactor internal material specifications identified in TR-31. In its response to RAI-TR31-001, Question 1, Part b, the applicant provided a table depicting the materials of construction for each of the reactor internal components. The staff evaluated this table and determined that the predominate materials of construction for all major core support structures, excluding bolting, are essentially limited to Types 304, 304H, 304L, and 304LN SSs. Furthermore, the staff confirmed the exceptions to the use of the 304-series SSs identified previously, specifically the use of nickel-based Alloy 690 for the clevis inserts, nickel-based Alloys 718 and 750 for the clevis insert-to-vessel bolts, nickel-based Alloy 750 for the irradiation specimen springs, and the addition of Stellite™ 6 hardfacing for wear surfaces on the radial keys, clevis inserts, and alignment pins. These exceptions were verified to be applicable only to these specific components. The high yield and tensile strength properties and corrosion resistance of nickel-based alloys justify their use for these specific components. These alloys are all acceptable in accordance with ASME Code, Section III. Therefore, the staff identified no safety-related issue associated with their use in these instances. DCD Revision 15, Section 4.5.1.3, previously addressed the qualification of Stellite™ 6 hardfacing for use in RCSs. The staff concludes that the applicant adequately addressed RAI-TR31-001, Question 1, Part b, because it provided a comprehensive list of material specifications for each reactor internal component that is consistent with DCD Revision 16, Section 4.5.2.1.

In RAI-TR31-002, Question 2, the staff requested that the applicant elaborate further on how it addressed the susceptibility of Type 304 and 304H SSs to various forms of corrosion and SCC, where welding on components fabricated using these materials could result in sensitization due to chromium depletion at grain boundaries. In its response to RAI-TR31-002, Question 2, the applicant reiterated its assertion that these SS grades have been used extensively for reactor internal components for currently operating Westinghouse plants. The applicant further stated that the available technical data on environmental degradation applicable to the currently operating Westinghouse plants are also applicable to the design of AP1000 reactor internal components. In addition, these materials have been assessed for a reactor internal component design life of 60 years, with respect to known mechanisms of IASCC and void swelling. DCD Revision 16, Section 4.5.2.1, states that internal material reliability programs are addressing the susceptibility to IASCC and void swelling in reactor internal components identified in the currently operating Westinghouse fleet, and the selection of materials for AP1000 reactor internal components considers information developed by these programs. TR-12, which was provided in WCAP-16620-P, Revision 0, "Consistency of Reactor Vessel Internals Core Support Structure Materials Relative to Known Issues of Irradiation-Assisted Stress Corrosion Cracking (IASCC) and Void Swelling for the AP1000 Plant," dated July 31, 2006, addresses the evaluation of AP1000 reactor internal components for potential susceptibility to IASCC and void swelling over the 60-year design life and the application of the IASCC and void swelling criteria established by the above material reliability programs. The staff concludes that the applicant's response to RAI-TR31-002, Question 2 was acceptable because it demonstrated that the applicant had adequately addressed the susceptibility of reactor internal components fabricated

from Types 304 and 304H SSs to known corrosion and SCC phenomena for reactor internal components.

Based on the above discussion, the staff concludes that the applicant provided an appropriate technical justification for the reactor internal material specification changes proposed in TR-31 because these proposed changes meet the requirements 10 CFR 50.55a and GDC 1 and will not adversely impact the safety of the AP1000 reactor design. Furthermore, the staff concludes that DCD, Section 4.5.2.1, fully represents these material changes. Therefore, the staff concludes that the DCD changes proposed by the applicant in TR-31 are acceptable.

The staff reviewed the proposed changes as they relate to Revision 16 of the AP1000 DCD. The AP1000 DCD incorporated the proposed changes identified in TR-31. Accordingly, these changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes contribute to the increased standardization of the certification information in the AP1000 DCD in accordance with 10 CFR 52.63(a)(1)(cii).

4.5.2.2 Conclusion

The staff finds that the changes to the material specifications for the reactor internal components proposed in TR-31 are technically acceptable because these changes meet the requirements of 10 CFR 50.55a and GDC 1 and will not adversely impact the safety of the AP1000 reactor design. Furthermore, the staff finds that the TR-31 conclusions regarding design changes related to the material specifications for AP1000 reactor internal components are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

4.5.3 Changes to In-Core Instrument Guide Tubes

In DCD Revision 17, the applicant proposed a change related to the replacement of in-core instrument guide tubes with QuickLoc assemblies, as well as several editorial modifications to DCD Tier 1, Section 2.1.3, "Reactor System."

4.5.3.1 Evaluation

The technical change involved the replacement of in-core instrument guide tubes with QuickLoc assemblies, as supported by APP-GW-GLE-016, "Impact of In-core Instrumentation Grid, Quicklocs and Changes to Integrated Head Package (IHP)." Also, this change included specifying the incore instrumentation QuickLoc assemblies as ASME Code Section III Classification and Seismic Category I classification in Tier 1 Table 2.1.3-1, and the QuickLoc assemblies as the pressure boundary components in Table 2.1.3-2. The staff reviewed APP-GW-GLE-016 and issued RAI-SRP15.4.8-SRSB-01 related to the rod ejection analysis in DCD Section 15.4.8. The staff determined the QuickLoc-related changes, as described in DCD Revision 17, Tier 1, Section 2.1.3, to be acceptable.

The first editorial modification is found in Table 2.1.3-2, which includes Design Commitment 2.a, and stipulates inspections, tests, analyses, and acceptance criteria (ITAAC) for the reactor upper internals rod guide arrangement. The applicant made an editorial change to refer to the correct figure describing the reactor upper internals rod guide arrangement. Specifically, Revision 15 listed Figure 2.3.1-1 instead of the correct figure (Figure 2.1.3-1).

The applicant made additional editorial modifications throughout Section 2.1.3 to correct the owrd order of "rod cluster control assemblies" (from "rod control cluster assemblies") and to clarify that the fuel assemblies are located in the containment location only after fuel loading (located in auxiliary building prior to fuel loading). The staff found these changes to be acceptable.

4.5.3.2 Conclusion

Based on the above evaluation, the staff concludes that the changes to DCD Tier 1, Section 2.1.3, are acceptable.

4.6 Functional Design of Reactivity Control Systems

The reactivity control systems for the AP1000 facility are the control rod drive system (CRDS), the reactor trip system, and the passive core cooling system, which can affect a safe shutdown, respond within acceptable limits during AOOs, and prevent or mitigate the consequences of postulated accidents.

4.6.2 Evaluation

In DCD Revision 17, Section 4.6, the applicant proposed the following revisions to Section 4.6.1, "Information for Control Rod Drive System": (1) a clarification that DCD Figure 4.2-8 provides the configuration of the driveline, including the CRDM, not the layout of the CRDS; (2) the deletion of the statement that the CRDM outer shroud is an integral portion of the head lifting system; and (3) the deletion of the "conduits for the in-core instrumentation" from the components located among the CRDM and supported by the integral head package.

The staff reviewed these revisions and concluded that they are editorial in nature. The applicant made the latter two revisions for the purpose of accuracy and consistency with the DCD Section 3.9.4.1.1 modifications to the integrated head package and redesign of the in-core instrumentation. These changes do not alter the functional design of the reactivity control systems and are, therefore, acceptable.

4.6.3 Conclusion

The NRC staff concludes that revisions to AP1000 DCD Section 4.6 are acceptable because the changes are editorial and do not alter the functional design of the reactivity control systems.

5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

5.1.1 Evaluation

The summary description of the "Reactor Coolant System and Connected Systems," of the AP1000 reactor coolant system (RCS) and connected systems, as well as their design bases, were evaluated by the staff of the U.S. Nuclear Regulatory Commission (NRC) in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design." Conforming changes to this section have been included in the revisions to the design control document (DCD) to reflect the design and bases changes to the reactor coolant pump (RCP), steam generators (SG), pressurizer, and normal residual heat removal system (RNS) designs that are discussed in Sections 5.4.1, 5.4.2, 5.4.5, and 5.4.7, respectively, as well as other DCD changes addressed in this chapter.

In Section 5.1 of the DCD, the applicant modified Figure 5.1-5, "Reactor Coolant System Piping and Instrumentation Diagram" (Sheet 1 of 3). The applicant stated that Figure 5.1-5 was modified by relocating the Loop 1 narrow-range and diverse actuation system (DAS) resistance temperature detector (RTD) upstream of the pressurizer surge nozzle, and by relocating the wide-range RTD upstream of the passive residual heat removal (PRHR) nozzle.

The applicant explained that the present Loop 1 location of the narrow-range device is a problem because pressurizer outsurges will cause erroneously high signals for some of the T-hot channels. The faster the RCS temperature is decreasing, the larger the resulting outsurge will be, and the higher the indicated Loop 1 temperature. The applicant concluded the most appropriate of several solutions is to relocate the narrow-range device upstream of the pressurizer surge nozzle. The staff agrees that the relocation of these RTDs would alleviate the influence of pressurizer outsurge and finds this to be acceptable.

In addition, AP1000 post-accident monitoring requirements also require that both wide-range T-hot RTDs be located at the top of the hot legs in order to detect voids. DCD Table 7.5-1, "Post-Accident Monitoring System," specifies two RCS wide-range T-hot measurements and one PRHR heat exchanger inlet temperature measurement. Therefore, the wide-range protection and safety monitoring system (PMS) hot leg RTD (TE-135A) needs to be relocated upstream of the PRHR nozzle to validate PRHR post-accident monitoring requirements. This location would also provide the desired direct post-accident reactor outlet temperature and PRHR inlet temperature. In conjunction with the Loop 2 hot leg wide-range RTD (TE-135B), the relocation of TE-135A satisfies the need for two wide-range T-hot and one PRHR inlet temperature for post-accident monitoring. In addition, the post-accident monitoring requirements also specify that both wide-range T-hot RTDs be located at the top of the hot leg in order to detect voids. Therefore, Note 24 is added to Figure 5.1-5 to specify that the thermal wells for the Loop 1 and Loop 2 wide-range RTDs (TE-135A and TE-135B) be located at the upper half of the hot leg. The staff finds this to be acceptable.

The information above was provided by the applicant in draft format but was not formally submitted to the staff. In a letter dated May 28, 2009, the applicant documented the details related to relocating the narrow-range and wide-range RTDs as discussed above which resolves this issue.

5.1.2 Conclusion

The staff has reviewed the proposed changes to Section 5.1, "Summary Description." Based on the evaluation described above, the staff concludes that the proposed changes to AP1000 DCD Figure 5.1-5 (Sheet 1 of 3) are acceptable pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 52.63(a)(1)(vii), "Finality of standard design certifications," on the basis that they contribute to the increased standardization of the certification information.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance with Codes and Code Cases

The staff approved Section 5.2.1.1, "Applicable Code Cases," of the AP1000 DCD, Revision 15, in the certified design. The applicant has proposed to make the following changes to Section 5.2.1 of the certified design:

5.2.1.1 Compliance With 10 CFR 50.55a

AP1000 DCD Tier 2, Section 5.2.1.1, "Code Compliance with 10 CFR 50.55a," uses the baseline code of the 1998 Edition throughout and including the 2000 Addenda for evaluations of the safety analysis and the AP1000 design certification (DC), except for 1989 Edition, 1989 Addenda for Articles NB-3200, NB-3600, NC-3600, and ND-3600 related to piping design. In a letter dated January 28, 2009, the applicant proposed changes to AP1000 DCD Tier 2, Section 5.2.1.1 to use 1989 Edition, 1989 Addenda for Articles NB-3210, NB-3620, NB-3650, NC-3620, NC-3650, ND-3620 and ND-3650. These proposed limitations represent only portions of Articles NB-3200, NB-3600, NC-3600, and ND-3600 disallowed by 10 CFR 50.55a(b)(1)(iii), "Codes and standards," for seismic design of piping. In request for additional information (RAI)-SRP5.2.1-EMB-03, the staff requested the applicant confirm whether the AP1000 piping design utilized the 1998 Edition, 2000 Addenda Article NB-3220 for the piping design. The staff also requested the applicant explain how the proposed changes would meet the requirements of 10 CFR 50.55a(b)(1)(iii), including the code requirements relevant to "Reversing Dynamic Loading in Piping," first introduced in the American Society of Mechanical Engineers (ASME) 1994 Addenda.

In the proposed changes to AP1000 Tier 2, DCD Section 5.2.1.1, the applicant indicated that it would use the 1989 Edition and the1989 Addenda sub-articles NB-3620, NC-3620, ND-3620, NB-3650, NC-3650 and ND-3650 for the seismic design of piping. The staff notes that these are the sub-articles describing the alternative provisions for seismic design of piping that were introduced in the 1994 Addenda Sections NB-3600, NC-3600, and ND-3600. The use of the above proposed sub-articles is consistent with the provisions of 10 CFR 50.55a(b)(1)(iii) and is, therefore, acceptable. Regarding the requirements in 10 CFR 50.55a(b)(1)(ii) relating to weld leg dimensions for socket welds, AP1000 piping design will comply with the requirements of 10 CFR 50.55a(b)(1)(ii) for socket weld dimensions, such that DCD Section 5.2.1.1 includes specific requirements including primary stress indices and stress intensification factor consistent with the requirements of 10 CFR 50.55a(b)(1)(ii). In its response dated April 1, 2009, to RAI-SRP5.2.1-EMB-03, the applicant stated that the AP1000 design does not utilize the alternative provisions introduced in NB-3200 from the 1994 Addenda for seismic design of piping. The applicant stated that AP1000 DCD Section 5.2.1.1 will be revised to include the use of the1989 Edition, 1989 Addenda for Subarticle NB-3220. The staff considers the applicant's response and the planned DCD changes to be acceptable. In a subsequent revision to the

AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

5.2.1.2 Applicable Code Cases

The staff approved Section 5.2.1.2, "Applicable Code Cases," of the AP1000 DCD, Revision 15, in the certified design. The applicant has proposed to make the following changes to Section 5.2.1.2 of the certified design:

1) DCD Tier 2 Section 5.2.1.2 was revised to reference Section 5.2.6.1, which includes a commitment that the combined license (COL) applicant will address consistency of the design with the construction practices (including inspection and examination methods) of the later ASME Code Edition and Addenda as well as Code cases approved subsequent to the DC.

5.2.1.2.1 Evaluation

The staff reviewed this change in accordance with Section 5.2.1.2, "Applicable Code Cases," of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants." The staff reviewed all changes identified by change marks in the AP1000 DCD. The staff did not re-review descriptions and evaluations in the AP1000 DCD, Revision 15, that were previously approved and that are not affected by the new changes.

The applicant revised DCD Tier 2, Section 5.2.1.2, to reference Section 5.2.6.1, which includes a commitment that the COL applicant will address consistency of the design with the construction practices (including inspection and examination methods) of the later ASME Code Edition and Addenda as well as Code cases approved subsequent to the DC. To ensure that appropriate Code cases are applied for inspection and examination, in RAI-SRP5.2.1-EMB01, the NRC asked the applicant to provide Code cases similar to those in DCD Table 5.2-3, which are applicable to regulatory guide (RG) 1.147, "Inservice Inspection Code Case Acceptability—ASME Section XI Division 1," and RG 1.192, "Operation and Maintenance Code Case Acceptability, ASME OM Code."

In a letter dated July 18, 2008, the applicant responded to RAI-SRP5.2.1-EMB-01 and indicated that the Code cases to be used for inservice inspection and testing should be included in the programs developed for these activities. The final safety analysis report (FSAR) in the COL application describes these programs. DCD Section 5.2.1.2 is not an appropriate place to identify the Code cases expected to be used for inservice inspection and inservice testing. However, the COL FSAR incorporates by reference the DCD statement that if Code cases other than those included in DCD Table 5.2-3 are used, a reconciliation review will be performed. It is important that the applicant provide the baseline Code cases similar to those in Table 5.2-3 that will be incorporated by reference in the COL application for use in inspection and testing. Without those baseline Code cases, the staff cannot determine whether the Code cases are the correct revision or if additional new Code cases are needed in the COL application to meet the requirements of 10 CFR 50.55a(a)(3), (b)(5), and (b)(6). The staff identified this as Open Item OI-SRP5.2.1-EMB-01.

In a letter dated July 15, 2009, the applicant stated that DCD Section 3.9.8.4 will be revised to require that the inservice test program identify the ASME Operation and Maintenance (OM) Code cases used. The applicant also indicated that DCD Section 5.2.1.2 would be revised to note that ASME Code, Section XI and the ASME OM Code and associated Code cases are not

directly used in the design of the reactor coolant pressure boundary. The ASME Code cases used in the inservice inspection and inservice testing programs would be determined as the COL information based on regulatory guidance and regulations, since RG 1.192 is not applicable to the AP1000 DC. In considering the applicant's planned changes to have Code cases used for the preservice and inservice testing included in the COL information, the staff finds the supplemental information in the letter dated July 15, 2009, to be reasonable and acceptable, based on the requirements in 10 CFR 50.55a. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In DCD Tier 2, Section 5.2.1.2, the applicant stated that the use of any Code case not approved in RG 1.84, "Design, Fabrication, and Materials Code Case Acceptability, ASME Section III," and RG 1.85, "Materials Code Case Acceptability—ASME Section III, Division 1," on Class 1 components is authorized as provided in 10 CFR 50.55a(a)(3) and the requirements of the DC. In RAI-SRP5.2.1-EMB-02, the staff asked the applicant to remove the reference to RG 1.85 from the DCD since the NRC withdrew it in 2003.

In a letter dated June 10, 2008, the applicant responded to RAI-SRP5.2.1-EMB-02 and noted that RG 1.85 is retained in the DCD for historical reasons. It also indicated that because the ASME Code used for the design of the AP1000 is the 1998 Edition, 2000 Addendum, RG 1.85 may include Code cases that are of interest. Since the last revision (Revision 31) of RG 1.85 includes Code cases with dates only up to 1994, the use of RG 1.85 is unrelated to meeting the requirements of the ASME Code, 1998 Edition through 2000 Addenda. Instead, the NRC withdrew RG 1.85 in 2003 but incorporated all Code cases from the original RG 1.85 into RG 1.84 and will continue to do so in its future updates. Therefore, the applicant should revise the DCD Tier 2 to RG 1.84 rather than RG 1.85 while updating the DCD Tier 2 to include the up-to-date information. The staff identified this as Open Item OI-SRP5.2.1-EMB-02.

In a letter dated June 17, 2009, the applicant indicated that DCD Section 5.2.1.2 would be revised to remove the references to RG 1.85. The applicant also indicated that the discussion of conformance to RG 1.84 in Appendix 1A would be revised from Revision 31 to Revision 32 to be consistent with the information in Table 1.9-1. The staff finds the applicant's response and the planned changes to the AP1000 DCD to be acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Revision 17 of AP1000 DCD included three new Code cases used in the AP1000 design, which were added to Table 5.2-3. These Code cases are N-655, "Use of SA-738, Grade B, for Metal Containment Vessels, Class MC, Section III, Division 1"; N-757, "Alternative Rules for Acceptability for Class 2 and 3 Valves, NPS 1 (DN25) and Smaller with Welded and Non-welded End Connections other than Flanges, Section III, Division 1"; and N-759-1, "Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Section III, Division 1." In RAI-SRP5.2.1-EMB-04, the staff requested the applicant confirm whether these Code cases were the most recent version and whether they were approved by the NRC in RG 1.84. If not, the staff requested that the applicant provide justification for using these Code cases in the AP1000 design in accordance with the requirements of ASME Code, Section III, NCA-1140.

In its response dated April 1, 2009, to RAI-SRP5.2.1-EMB-04, as supplemented by a letter dated June 8, 2009, the applicant stated that ASME Section III Code Cases N-757 and N-759-1 are not included in Revision 34 of RG 1.84. The staff notes that Code Case N-655 was

conditionally approved by the NRC in Revision 33 of RG 1.84. The applicant also indicated that the AP1000 design would apply the latest Code Cases N-655-1, N-757-1 and N-759-2 as identified in its letter dated June 8, 2009. The applicant provided justification required by 10 CFR 50.55a for the use of these Code cases as discussed in the following paragraphs.

See Section 3.8.2.5 of this report for the evaluation of ASME Code Case N-655. Since Code Case N-655-1 revised Code Case N-655 by replacing Supplementary Requirement S-17 with Supplementary Requirement S-1, and is considered to be equivalent with respect to controlling the quality of the SA-738 material, the staff finds that Code Case N-655-1 provides an acceptable level of quality and safety and is authorized pursuant to 10 CFR 50.55a(a)(3)(i). The applicant indicated that DCD Table 5.2-3 would be revised to include Code Case N-655-1 in lieu of N-655. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Code Case N-757 allows the use of alternate rules for the design of instrument, control and sampling line valves, Class 2 and 3 NPS 1 (DN25) and smaller, with welded and non-welded end connections other than flanges. The ASME Code requirements for the design of these valves are in Articles NC-3500 and ND-3500 for ASME Code Class 2 and 3 valves. The standard design rules in Paragraphs NC-3512 and ND-3512 require that the minimum wall thickness satisfy the thickness requirements specified in the valve standard ASME B16.34, "Valves- Flanged, Threaded, and Welding End." Paragraphs NC-3513 and ND-3513 provide the alternate design rules that may be used in place of NC-3512 and ND-3512 when permitted by the design specification. However, these alternate rules only apply to valves with butt welding end connections and socket welding end connections. Code Case N-757-1 allows the use of the alternate design rules for welded and non-welded end connections other than flanges, in the design of small valves. Code Case N-757-1 states that these valves may meet the design requirements of Section III, Division 1, Class 2 and 3 rules in Paragraphs NC-3512 and ND-3512 provided specific additional requirements are met. Based on a discussion in a conference call with the staff on June 19, 2009, for the use of Code Case N-757-1, the applicant addressed, in a letter dated July 2, 2009, issues regarding the operating experience of non-welded valves identified in NRC Information Notice (IN) 84-55, "Seal Table Leaks at PWRs," and IN 92-15, "Failure of Primary System Compression Fitting." The applicant stated that information and cautions would be provided in design documents (e.g., design specifications and instruction manuals) including: (a) not mixing the parts from one manufacturer to another; (b) following manufacturer's recommended instructions for installing compression fittings; and (c) providing training and procedures for personnel performing the work. The staff concludes that the use of Code Case N-757-1 will provide an acceptable level of quality and safety for the design of instrument, control and sampling line valves, Class 2 and 3 NPS 1 (DN25) and smaller, with welded and non-welded end connections other than flanges, and is, therefore, authorized pursuant to 10 CFR 50.55a(a)(3)(i). The applicant indicated that DCD Table 5.2-3 would be revised to include Code Case N-757-1 in lieu of N-757. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Code Case N-759-2, "Alternative Rules for Determining Allowable External Pressure and Compressive Stresses for Cylinders, Cones, Spheres, and Formed Heads, Section III, Division 1," is intended for use in the design of AP1000 to address an issue with the primary side depressurization transients. Code Case N-759-2 provides an alternative methodology for the SG tube collapse analysis based on theoretical buckling equations and buckling tests on fabricated cylindrical tubes.

In a letter dated July 2, 2009, the applicant provided supplemental information stating that the use of theoretical buckling equations and buckling tests provides assurance that Code Case N-759-2 would provide an acceptable level of quality and safety. The applicant also stated that compliance with the existing rules of the ASME Code without the use of Code Case N-759-2 would require a re-design of the AP1000 SG tube bundle. The staff finds that the use of Code Case N-759-2 provides an acceptable level of quality and safety and is, thus, authorized pursuant to 10CFR 50.55a(a)(3)(i). The applicant indicated that AP1000 DCD Tier 2, Table 5.2-3 would be revised to replace Code Case N-759-1 with N-759-2. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Revision 17 to the AP1000 DCD, Introduction, Table 1-1, "Index of AP1000 Tier 2 Information Requiring NRC Approval for Change," includes Tier 2* information requiring NRC approval for change. AP1000 DCD Tier 2, Table 5.2-3 also includes Tier 2* information as specified in the footnotes to the table. In RAI-SRP5.2.1-EMB-05, the staff requested the applicant revise Table 1-1 to be consistent with Table 5.2-3. In its response to RAI-SRP5.2.1-EMB-05, the applicant indicated that it had identified inconsistencies and an omission in Table 1-1 and planned to make several changes. For example, the reference to a specific Code case would be expanded to cover all ASME Code cases with a reference to Table 5.2-3. A note specifying that the 2001 Edition of the ASME Code, Section III, including 2002 Addenda, applies to containment design would be included in Table 1-1. The table would also be revised to remove incorrect references to Sections 3.8.2.5 and 5.2.1.1 for this item. An item would be added to Table 1-1 listing the baseline ASME Code Edition and Addenda with reference to Section 5.2.1.1. The staff considers the applicant's planned changes to the DCD provide consistency in the applicable DCD sections and are, therefore, acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

By letters dated April 28 and June 8, 2009, the applicant requested the use of Code Case N-782, "Use of Code Editions, Addenda, and Cases Section III, Division 1," for the AP1000 design. This code case is not included in RG 1.84, Revision 34. As required by 10 CFR 50.55a(a)(3), the applicant requested approval for the use of this ASME Code case as a proposed alternative to the rules of 10 CFR 50.55a.

Code Case N-782 provides that the Code Edition and Addenda endorsed in a design certified by the regulatory authority may be used for systems and components constructed to ASME Code, Section III requirements. This Code case updates Paragraph NCA-1140 of ASME Code, Section III to address the licensing process under 10 CFR Part 52, "License, certifications, and approvals for nuclear power plants." The applicant indicated that the use and approval of Code Case N-782 was needed to align ASME Code requirements to the ASME Code Edition and Addenda cited in AP1000 DCD Tier 2, Section 5.2.1.1. A reference to Code Case N-782 would be included in component and system design specifications and design reports to permit certification of these specifications and reports to the Code Edition and Addenda cited in the DCD and approved by the NRC. The applicant indicated that the use of Code Case N-782 facilitates the use of the ASME Code Edition and Addenda included in the AP1000 DC. Therefore, it would provide the same level of quality and safety as was included in the information reviewed for the AP1000 DC. The information provided in the applicant's letter is generic and applies to all COL applicants referencing the AP1000 DC. The staff concludes that the use of Code Case N-782 provides an acceptable level of quality and safety and is, therefore, authorized pursuant to 10 CFR 50.55a(a)(3)(i). AP1000 DCD Tier 2, Table 5.2-3 will

be revised to reference Code Case N-782. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

5.2.1.2.2 Conclusions

Based on the information provided in the applicant's responses to the RAIs, the staff finds that the AP1000 DCD amendment conforms to the guidance provided in RG 1.206, "Combined License Applications for Nuclear Power Plants." This satisfies the requirements of 10 CFR 50.55a and General Design Criteria (GDC) 1, "Quality Standards and Records," in Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic licensing of production and utilization facilities," and, therefore, is acceptable.

5.2.3 Reactor Coolant Pressure Boundary Materials

5.2.3.1 Evaluation of Zinc Addition to the Reactor Coolant

5.2.3.1.1 Summary of Technical Information

In Revision 16 to the AP1000 DCD, the applicant proposed design changes to incorporate the ability to inject zinc into AP1000 reactor coolant. In a letter dated April 5, 2006, the applicant submitted TR-32, "Zinc Addition," APP-GW-GLN-002, to provide technical justification for the proposed zinc addition into the reactor coolant.

The request consists of a modification to the AP1000 design to incorporate the ability to inject a small quantity of zinc acetate into the RCS. Operation with chemical zinc in the cooling system has been shown to change the oxide film on primary components, reducing occupational radiation exposure and the potential for crud formation, crud deposition on the fuel rods, and subsequent power shifts. The applicant proposed to provide zinc addition as an optional mode of operation.

Implementation of the requested zinc injection capability in the AP1000 results in the following Tier 2 changes to the DCD:

1. Add note 7 in Table 5.2-2, "Reactor Coolant Water Chemistry Specifications," to specify the maximum zinc concentration.

2. In Table 6.2.3-1 revise the chemical and volume control system hydrogen injection to the RCS line isolation device to be normally open.

3. Revise the third paragraph of Section 9.3.6.2.1, "Purification," to the effect that the mixed bed demineralizers will also remove zinc.

4. Add Section 9.3.6.2.3.3, "Zinc Addition," to the effect that zinc may be added.

5. Revise Figure 9.3.6-1 to include: Valve V092 is changed to normally open, a reducer is added downstream of V065, the portion of the H2/ZINC ADD line from the reducer to the return line is renumbered as L064, and the specification of L064 is changed to .5" BBC.

5.2.3.1.2 Evaluation

GDC 4, "Environmental and Missile Dynamic Effects Design Bases," requires that structures, systems, and components (SSCs) important to safety shall be appropriately protected against environmental and dynamic effects. The staff reviewed changes related to this section to ensure the compatibility of components with the environmental conditions created by the addition of zinc.

In letters dated August 1, 2006, and September 28, 2006, the staff requested additional information regarding the effects of zinc injection. The applicant responded in letters dated September 8, 2006, and December 12, 2006. The staff reviewed the responses and included that information in the overall evaluation, which follows.

Zinc acetate will be added using the same piping and valving as the hydrogen addition. The proposed hardware change is to replace a portion of the 2.5 centimeter (cm) (1-inch (in)) pipe (downstream from the containment isolation valve) with a heavier wall 1.3 cm (½-in) pipe. This will reduce the piping volume and substantially reduce the transit time for the hydrogen and the zinc acetate injected material. The integrity of the pressure boundary is not affected. Both hydrogen and zinc acetate injections have low transit velocities; thus, flow stability is not a problem. The containment isolation function signal, the containment isolation, the valve designation as active (Table 3.9-12 in the DCD), the safety-related mission, the inservice testing type and frequency requirements (Table 3.9-16), and the valve functional requirements for containment isolation (Tier 1, Table 2.3.2-1) are not affected.

The staff concludes that zinc addition in the primary coolant reduces radiation fields and the formation of crud, which may result in increased personnel exposure and in axial power shifts, respectively. Regarding the effect of zinc in reducing primary water stress-corrosion cracking, the applicant clarified that it no longer takes credit for the mitigation of this type of cracking based on zinc addition. The applicant believes that there is sufficient margin in the selection of new materials and that credit from zinc addition is not needed. The staff finds that the presence of zinc in the primary water will not cause aging-related degradation; therefore, it is acceptable.

The staff questioned the effect of the thin oxide film that forms in the presence of zinc with respect to the potential increase of the heat transfer coefficient and the potential increase in fuel and cladding operating temperature. The applicant submitted additional information in the form of cladding oxidation results that demonstrated that existing plant operational data (using ZIRLO™ cladding) did not exhibit increased oxidation, suggesting that the cladding is operating at comparable temperatures. The staff finds that the presence of zinc meets the requirements of GDC 4 and does not decrease the cladding-to-coolant heat transfer coefficient; thus, the staff finds the presence of zinc in the primary water to be acceptable. These changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant.

5.2.3.1.3 Conclusions

On the basis of its review of TR-32, the staff finds that the requested modification for zinc addition to the primary water meets the requirements of GDC 4 and is acceptable for ZIRLO™ fuel cladding. Furthermore, the staff finds that the TR-32 conclusions regarding the evaluation for zinc addition to the primary water are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are

acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.2.3.2 Evaluation of Reactor Coolant Pressure Boundary Materials

5.2.3.2.1 Summary of Technical Information

In Revision 16 to the AP1000 DCD, the applicant proposed design changes related to the AP1000 pressure boundary materials. In a letter dated May 24, 2007, the applicant submitted TR-33, "Pressure Boundary Material Change" APP-GW-GLN-009, Revision 1, which provides technical justification for the design changes related to the AP1000 pressure boundary materials. The pressure boundary materials selection appears in DCD Tier 2, Section 5.2, Table 5.2-1. The pressure boundary materials changes identified in TR-33 are: (1) the revision of some material designators to be consistent with the ASME Code of Record for the RCS pressure boundary components, which is the 1998 Code and 2000 Addenda; (2) the correction of discrepancies in the AP1000 DCD in the specification of materials for some components; (3) the addition of materials to DCD Table 5.2-1 to address previously identified issues, to provide fabrication flexibility, and to ensure adequate material supply; (4) the relaxation of the maximum copper limit allowed in the reactor vessel beltline forging and weld material to reduce schedule risk and cost while maintaining the required performance; and (5) the relaxation of the maximum delta ferrite limit in weld materials to reduce schedule risk and cost while maintaining the performance requirements.

Revision 17 revised Section 5.2.3.1 of the AP1000 DCD, Tier 2, to include ASME Code Filler Metal Specifications SFA 5.1 and 5.17 for carbon steel, ASME Code Filler Metal Specification SFA 5.22 for stainless steel which allows the use of fluxed-cored filler metal to be used for welding the root pass, and ASME Code Filler Metal Specification SFA 5.30 for consumable inserts. Revision 17 also revised AP1000 DCD, Table 5.2-1, to allow an option to use carbon steel (SA-508, Class 1A) instead of alloy steel (SA-508, Grade 3, Class 2) on the pressure forgings (including nozzles and tubesheets for the SG), and allows an option to use carbon steel (SA-508, Grade 1) instead of stainless steel (SA-336, Grades F304, F304L, F304LN, F316, F316L and F316LN) on the pressure forgings for the RCP. In addition, Table 5.2-1 of the DCD was changed in Revision 17 to add material specification SA-338 for the pressurizer nozzle safe-ends, and to replace material specifications SA-312 and SA-376 for seamless pipe with material specification SA-479 for hot-rolled or cold-rolled bar stock with a unified numbering system (UNS) designation S21800.

The staff reviewed information in the AP1000 TR-33 that supports Revision 16 to the AP1000 DCD, along with the associated changes made in Revision 17. The staff's findings are summarized below.

5.2.3.2.2 Evaluation of Revision 16 Changes

This evaluation addresses the impact of the changes identified in TR-33 related to the Class 1 pressure boundary materials, in such areas as the reactor vessel and internals, pressurizer, and SGs.

a. ASME Code of Record Update

Reactor vessel components (head plates, shell courses, shell flange, and appurtenances to the control rod drive mechanism (CRDM) material designators) were revised to be consistent with

the ASME Code of Record for the RCS pressure boundary components, which is the 1998 Code and 2000 Addenda. These changes have no impact on the safety evaluation performed by the staff because the materials identified continue to meet the requirements of the ASME Code, Section III. Hence, the changes are acceptable.

b. Material Addition

Reactor vessel components (appurtenances to the CRDM, instrumentation tube appurtenances, and monitor tubes) materials have been added. The added materials have been used in pressurized-water reactors (PWRs) in the past. In addition, reactor vessel components (nozzle safe ends, appurtenances to the CRDM, instrumentation tube appurtenances, upper head, monitor tubes, and vent pipe materials) currently identified in the DCD and TR-33 include corresponding class, grade, or type. The identified added materials and the additional information on grades, types, or classes meet the requirements of the ASME Code, Sections II and III in accordance with the guidance of NUREG-0800 Section 5.2.3. Therefore, the inclusion of the identified materials and the additional information on grades, types, or classes presented in TR-33 have no impact on the conclusions reached by the staff in its review of the AP1000 DCD, Revision 15. Hence, the additions are acceptable. Section 4.5.1 of this report discusses the evaluation of the material changes for the CRDM components.

c. Reactor Vessel Beltline Forging and Weld Chemical Composition

The current AP1000 DCD, Revision 15, specifies maximum 0.03 weight-percent (wt%) copper (Cu) for reactor vessel beltline forgings and welds. The copper limits were established to address irradiation embrittlement concerns. TR-33 proposed that the copper limit for the reactor vessel beltline forging and weld material be 0.06 wt%. Forgings with specified maximum copper limits of 0.03 wt% are outside the current practice for the potential forging suppliers since most specifications are commonly in the range of 0.05 to 0.1 wt%.

Changing the wt% of copper impacts the end-of-life (EOL) reference temperature-pressurized thermal shock (RT_{PTS}), which is calculated according to the methodology prescribed in 10 CFR 50.61(c)(1), "Fracture toughness requirements for protection against pressurized thermal shock events." However, even though the RT_{PTS} value is increased, there is still substantial margin from exceeding the screening criteria limit set forth in 10 CFR 50.61(b)(2). Table 5-1 summarizes the impact of increased copper on the EOL RT_{PTS} (using the current RT_{PTS} criteria).

Table 5-1. Impact of Increased Copper on the EOL RT_{PTS}

	EOL RT_{PTS} CU =.03 wt% Current Limit	**EOL RT_{PTS} CU = 0.06 wt% Proposed Limit**	**EOL RT_{PTS} Screening Criteria**
Beltline Forging	18.9 °Celsius (C) (66 °Fahrenheit (F))	34.4 °C (94 °F)	<132.2 °C (270 °F)
Beltline weld	36.7 °C (98 °F)	64.4 °C (148 °F)	< 148.9 °C (300 °F)

Based on the data shown in Table 5-1, the impact of higher copper content on the beltline weld and beltline forging EOL RT_{PTS} is small and does not challenge the screening criteria of 10 CFR 50.61(b)(2).

Adjusted reference temperature (ART) will be slightly increased because of the higher copper content and that will decrease the allowable operating temperature. However, this will not result in any significant restrictions on plant operations. The pressure-temperature (P/T) curve changes will not significantly affect the low-temperature overpressure protection (LTOP) system evaluation and the resulting parameters for the normal RNS relief valve.

Thus, the impacts of the increase in copper content of the beltline weld and beltline forging are insignificant. Taking into consideration the increase in copper content, the staff's review finds that the requirements of GDC 1 and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," and 10 CFR 50.55a regarding quality standards are met by compliance with the provisions of the ASME Code, Section III, for the beltline forgings and weld materials. The materials also meet the requirements of Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50 and 10 CFR 50.61. Therefore, the staff's review finds the changes to the reactor vessel beltline forging and weld chemical compositions acceptable.

d. Delta Ferrite Limits

AP1000 DCD Revision 15, specifies an upper delta ferrite limit of 13 ferrite number (FN). The proposed change is to increase the upper-shelf limit to 20 FN and to clarify the acceptable methods to verify the delta ferrite content. The proposed change to the maximum delta ferrite content is 20 FN for filler metal compositions with low molybdenum contents and 16 FN for weld filler materials with higher molybdenum content, such as Types 316/316L. The increase in maximum allowable delta ferrite levels will increase the availability of suppliers and flexibility in fabrication, resulting in a decrease in cost and fabrication time without compromising material performance.

The upper-shelf limit of 20 FN is still within the guidelines established by the staff under RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," which states that weld pad test results showing an average FN from 5 to 20 indicate that the filler metal is acceptable for production welding of Class 1 and 2 austenitic stainless steel components and core support structures. In addition, the upper- and lower-shelf limit FN content is within the acceptance standards under ASME Code, Section III, NB-2000. Based on the discussion above, the change is within the limits prescribed by the staff and the ASME Code and is acceptable.

e. Primary and Auxiliary Piping

Section 3B.2, "Potential Failure Mechanisms for AP1000 Piping," of Appendix 3B, "Leak-Before-Break Evaluation of the AP1000 Piping," to the AP1000 DCD, Revision 15, states that SA312TP316LN- and SA304TP304L-grade steels selected for primary and auxiliary piping are both resistant to intergranular stress-corrosion cracking and wall thinning due to erosion-corrosion effects. The proposed change replaces the references from specific materials to a more generic statement, which calls for Series 300 stainless steel materials. The proposed change also states that these materials were chosen because of their proven operating experience in low- or no-oxygen environments with no incidents for a number of years. The proposed change also states that RG 1.44, "Control of the Use of Sensitized Stainless Steel," "will be used to maintain experiences of the PWR applications for the use of Series 300 stainless steel materials." The applicant also revised Table 5.2-1 of DCD Section 5.2 to include the specific material Types 304, 304L, 316, and 316L, in addition to the current Types 304LN and 316LN used for the reactor coolant pressure boundary components, including piping.

In proposing this change, higher carbon stainless steel materials (Types 304 and 316) could be used in the reactor coolant pressure boundary, including piping, and may be more susceptible to sensitization due to heat treatments or welding. However, for the applications proposed, the reactor coolant environment to which these materials will be exposed is either a low- or no-oxygen environment in part because of the use of oxygen scavenger chemicals such as gaseous hydrogen and hydrazine. Past operating experience with the use of Series 300 stainless steels in such low- or no-oxygen environments has not shown any significant challenges to piping integrity for a number of years. Therefore, the staff concurs that the use of Series 300 stainless steels is the most proven application in the fleet, taking into consideration operational experience. In addition, the DCD mentions RG 1.44 as the guideline followed in the selection and use of the material and the control methods to avoid sensitization. Therefore, the staff finds that the use of these stainless steel materials exposed to PWR reactor coolant water is acceptable. However, the staff notes that the use of these materials on any piping that implements leak-before-break will be verified as part of inspections, tests, analyses, and acceptance criteria (ITAAC) during the COL phase using the actual material properties and final, as-built piping analysis to ensure that the piping using these materials still meets the leak-before-break bounding analysis curves (which were originally evaluated for Types 304LN and 316LN piping material).

f. Steam Generators

TR-33 revises the material designations for pressure boundary components to be consistent with the ASME Code of Record (1998 Code with 2000 Addenda).

Table 5.1-2 of the DCD updates material class and grade designators. For example, the SG tubing material designator was revised from SB-163 TP 690TT to SB-163 N06690. ASME Specification SB-163 identifies the chemical and physical properties of the material and notes that the material is annealed. Section 5.2.3.1.1 of this report states that the SG tubes are made of thermally treated Alloy 690 material. Section 5.2.3.1, "Material Specifications," of the DCD Revision 15 also states that the SG tubes use Alloy 690 in the thermally treated form.

The material designators for the SG manway closure studs and nuts were also updated to reflect the more correct way to identify bolting material. The studs are SA-193 Gr. B-7 and the nuts are SA-194 Gr. 2H.

The staff finds that the revisions and updates to the DCD related to the SG components do not alter the staff's conclusions in NUREG-1793 and are consistent with NUREG-0800 Section 5.2.3. Furthermore, the requirements of GDC 1, GDC 30, "Quality of Reactor Coolant Pressure Boundary," and 10 CFR 50.55a regarding quality standards are met for material specifications by compliance with the applicable provisions of ASME Code, Section III. Therefore, the staff finds these changes to the DCD related to SG components acceptable.

5.2.3.2.3 Evaluation of Revision 17 Changes

a. Use of Carbon Steel

Revision 17 to the AP1000 DCD proposed the use of carbon steel as an option for reactor coolant pressure boundary components, which include the SG and the RCP. Therefore, the staff requested the applicant provide justification and operating experience on the use carbon steel or remove the use of carbon steel in reactor coolant pressure boundary components.

In its December 31, 2009 response to RAI-SRP5.2.3-CIB1-01, Revision 1, the applicant stated that carbon steel is used only for those pressure boundary components which are not normally exposed to the reactor coolant (i.e., RCP components - stator main flange, stator shell and external heat exchanger supports). These components use carbon steel in lieu of 304 stainless steel due to the stresses induced by the studs and the main flange. Specifically, a 304 stainless steel main flange may experience an increase in the main flange stud load at operating conditions due to the thermal expansion between the steel studs and the 304 stainless steel main flange. Therefore, the carbon steel will provide a higher margin against stud/nut to main flange bearing stresses. In addition, Note 4 is proposed to be added to Table 5.2.3-1 of the AP1000 DCD to state the use of carbon steel base material and weld material are limited to only these components of the RCP. In addition, in its April 7, 2010 response to RAI-SRP5.2.3-CIB1-01, Revision 2, the applicant stated that carbon steel will not be used for the SG reactor coolant pressure boundary forgings, and proposed to remove the carbon steel material from Table 5.2-1. The staff notes that carbon steel is only used in 3 components for the RCP that is not in contact with the reactor coolant. Therefore, the staff finds the use of carbon steel for only non-wetted components of the RCP acceptable, and that there is no other use of carbon steel for the reactor coolant pressure boundary. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

b. Use of Flux-cored Welding in Root Pass

Revision 17 to the AP1000 DCD included the use of ASME Code Filler Metal Specification SFA 5.22, which allows the use of fluxed-cored filler metals to be used for welding the root pass in all stainless steel reactor coolant pressure boundary components. The staff has concerns that the use of flux-cored filler metal in the root pass may introduce slag inclusions in the root weld layer in contact with the reactor coolant, thereby providing a crack initiation site. In its August 12, 2009 response to RAI-SRP5.2.3-CIB1-01, Revision 0, the applicant stated that the use of flux bearing weld processes for root welds in which the root layer is exposed to the reactor coolant are prohibited unless the backside is first back-gouged to remove the root layer and re-welded from the backside. In addition, in its April 7, 2010 response to RAI-SRP5.2.3-CIB1-01, Revision 2, the applicant proposed Notes 3 and 8 to Table 5.2-1 of the AP1000 DCD, which specifies this prohibition of using flux-bearing filler metal for weld root passes. The NRC staff finds the use of fluxed-cored filler metals with the addition of Notes 3 and 8 to Table 5.2-1 of the AP1000 DCD acceptable since it prohibits the use of flux containing weld processes on the root layer, which would be in contact with the reactor coolant, thereby ensuring the weld integrity by minimizing the occurrence of a crack initiation site in the root layer. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

c. Use of Consumable Inserts

Section 5.2.3.1 of the AP1000 DCD, Revision 17 proposed the use of ASME Code Filler Metal Specification SFA 5.30 for consumable inserts for stainless steel. In addition, Table 5.2-1 of the AP1000 DCD, Revision 17 included Alloy 690 safe ends for the SG nozzle. However, Revision 17 did not include ASME Code Filler Metal Specification SFA 5.30 for consumable inserts joining the SG Alloy 690 safe ends to the RCP casing to ensure full penetration of the weld joint. In its response to RAI-SRP5.2.3-CIB1-01, Revision 3, dated July 9, 2010, the applicant stated that welding of the SG Alloy 690 safe end to the RCP casing is performed from both sides to achieve full penetration. The weld design features a double-sided weld since there is access to the second side for back-gouging and re-welding. Therefore, the staff finds

the addition of SFA 5.30 acceptable for providing a weld joint to achieve full penetration, particularly for installation closure welds, and the use of a double-sided weld for the SG to the reactor coolant weld to ensure full penetration, which is in accordance with Section III of the ASME Code.

d. RCP – Use of Carbon Steel and the RCP Flywheel Analysis

Since carbon steel will be utilized in the RCP, the staff reviewed how this would affect the flywheel analysis described in Section 5.4.1.4 of this report. In its August 12, 2009 response to RAI-SRP5.2.3-CIB1-01, Revision 0, and its December 31, 2009 response to RAI-SRP5.2.3-CIB1-01, Revision 1, the applicant stated that carbon steel is only used for the stator main flange, stator shell and the external heat exchanger supports. The staff determined that since the flywheel analysis described in Section 5.4.1.4 only credits the RCP casing, thermal barrier, stator closure ring and the stator lower flange for containing the flywheel, the use of carbon steel has no impact on the flywheel analysis.

However, to clarify the use of the carbon steel base material and welding filler metal for the RCP, the applicant stated in response to RAI-SRP5.2.3-CIB1-01, Revision 1, that full penetration weld joints are used for welding the stator shell (carbon steel) to the stator main flange (carbon steel) and the stator lower flange (stainless steel). The stator shell (carbon steel) to the stator lower flange (stainless steel) is a dissimilar metal weld. The stator shell is buttered with ASME Code Filler Metal Specification 5.9, Classification ER309, and then the weld joining the lower flange to the ER309 buttering uses ASME Code Filler Metal Specification 5.9, Classification 308. Both welds have radiographic inspection requirements pursuant to Section III of the ASME Code, and are surface examined on both the inside and outside diameter surfaces. The staff finds the welding and inspections for the carbon steel components for the RCP will be performed in accordance with the criteria of Section III of the ASME Code for full penetration to ensure the integrity of the welded components.

However, the NRC staff noted that since Filler Metal Specification SFA 5.9, Classification ER309 is used for the weld buttering on the RCP stator shell, which is subsequently post-weld heat treated, the ER309 buttering may become sensitized when exposed to the post-weld heat treatment. In its response to RAI-SRP5.2.3-CIB1-01, Revision 2, dated April 7, 2010, the applicant proposed to add Note 5 to Table 5.2-1 of the AP1000 DCD to clarify the maximum carbon content for stainless steel welds, which would be exposed to post-weld heat treatment that may cause sensitization. In its response to RAI-SRP5.2.3-CIB1-01, Revision 3, dated July 9, 2010, the applicant further clarified the carbon content of stainless steels, which meets the guidance in RG 1.44. Therefore, since the maximum carbon content is specified to prevent sensitization and the guidance of RG 1.44 is used when post-weld heat treating stainless steel, the staff finds Note 5 acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

e. Use of EC and EQ

In addition, the applicant's letter dated April 7, 2010, proposed new changes to Table 5.2-1 of the AP1000 DCD to add strip electrodes (EQ) for SFA 5.9 filler materials, and composite cored/stranded electrodes (EC) for SFA 5.9 and SFA 5.14 filler materials from ASME Code, Section II. EQ filler materials currently listed in SFA 5.9 and SFA 5.14 of ASME Code, Section II are used for cladding purposes, and have been used in current operating plants. The use of this filler material allows increase weld surface area deposition, which is particularly

beneficial when performing weld cladding of vessels. In its response to RAI-SRP5.2.3-CIB1-01, Revision 3, the applicant provided additional information on the use of EC filler material and clarified the use of EQ and EC in Table 5.2-1 of the AP1000 DCD. The applicant stated that EC filler material can be in the form of stranded wire and metal cored wire. The stranded wire is used primarily for automatic gas-tungsten arc welding since it is more flexible, which aids in smooth and consistent wire feeding into the weld puddle. The staff notes that EC filler material provides a more consistent and repeatable weld, which aids in minimizing welding defects, thereby improving the integrity of the weld. Therefore, the staff considers the use of EC filler materials specified in the Section II of the ASME Code acceptable.

The applicant also stated in its response to RAI-SRP5.2.3-CIB1-01, Revision 3, that the metal cored composite filler metal, EC, is used with the gas metal arc welding (GMAW) process, and due to the higher current density of these filler materials, results in higher weld deposition, and can be better controlled by the welder, thereby enabling acceptable welds to be produced consistently. The EC filler material contains no flux, and the mechanical and chemical properties of the resulting weld are identical to other filler materials used for the GMAW process. Therefore, the staff finds the use of EC filler material specified in SFA 5.9 of the ASME Code, Section II, acceptable since these filler materials produce welds that have the same chemical and mechanical properties as other SFA 5.9 filler materials allowed by the ASME Code. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

f. Use of 52M

In a letter dated April 7, 2010, the applicant proposed new changes to Section 5.2.3.1 and Table 5.2-1, Note 7, to include the use of ASME Code Filler Metal Specification 5.14, Type ENiCrFe-7A (UNS N06054) SFA 5.14. In its response to RAI-SRP5.2.3-CIB1-01, Revision 3, the applicant clarified that the use of the nickel alloy weld material in SFA 5.14, which includes UNS N06054 (Alloy 52M) would improve weldability. The NRC staff notes that N06054 filler material is more resistant to ductility dip cracking than UNS N06052 and W86152 filler materials, thereby ensuring the integrity of the weld. In addition, other filler material in this classification have been, and are currently being developed to further improve the weldability by minimizing the potential for both hot cracking and ductility dip cracking (microfissuring). Therefore, the staff agrees that the use of these filler material alloys developed for improved weldability for SFA 5.14, which are included in ASME Code, Section II, can be used since these filler materials provide an acceptable level of quality to ensure the integrity of the weld. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

g. Pressurizer Safe-ends

Material Specification SA-338, Grades F316, F316L and F316LN was added in Revision 17 of AP1000 DCD, Table 5.2-1 for the pressurizer safe-ends. However, the NRC notes that this material specification does not exist in ASME Code, Section II. In its response to RAI-SRP5.2.3-CIB1-02, dated December 11, 2009, the applicant changed the material specification to SA-336 to correct the editorial error. The staff finds the addition of Material Specification SA-336 for stainless steel acceptable for use on the pressurizer safe-ends since it is in Section II of the ASME Code and has acceptable operating experience in current plants. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

h. Use of Quickloc and SA-479

Table 5.2-1 of the AP1000 DCD was changed in Revision 17 to replace Material Specifications SA-312 and SA-376 (seamless pipe) with Material Specification SA-479 for hot-rolled or cold-rolled bar stock with no supporting justification or operating experience. In addition, the UNS designation S21800 for Material Specification SA-479 has higher carbon content than Types 304 and 316, along with lower chromium content with no molybdenum additions, thereby making the material susceptible to sensitization if welded. Therefore, the staff requested a discussion on the compatibility of this material with the reactor coolant, along with operating experience. In its response to RAI-SRP5.2.3-CIB1-01, Revision 1, the applicant stated that Material Specification SA-479 replaced SA-312 and SA-376 because the AP1000 design incorporated a Quickloc mechanism to replace the previous incore instrument thimble assemblies (IITAs). The new design includes eight Quickloc penetrations in lieu of the 42 individual IITA penetrations on the reactor vessel head. The applicant also provided operating experience for this material, which includes:

- Used for Quicklocs on several operating plants (Waterford 3, St. Lucie 1 and 2, and Calvert Cliffs 1 and 2) starting from the mid 1990's.

- Used for core exit nozzle assembly pressure boundary parts since 1988 in approximately 50 plants worldwide.

- Used for various reactor vessel internal parts.

The Material Specification SA 479, UNS 21800 (also known as Nitronic 60) is procured in the solution annealed condition, and is used primarily for areas to minimize wear, galling and fretting due to the additions of silicon and manganese. The staff notes that although Nitronic 60 has a maximum carbon content of 0.10 percent versus 0.08 percent for Types 304 and 316 stainless steels, sensitization of the Nitronic 60 is not a concern for the Quickloc since it is provided in the solution annealed condition and is not subject to welding. In addition, intergranular corrosion tests are performed in accordance with the American Society for Testing and Materials (ASTM) A262-02a, Practice E, as outlined in RG 1.44, to ensure the material is not sensitized. The staff finds that the Nitronic 60 material has satisfactory operating experience in contact with the reactor coolant since the mid 1980's and, therefore, finds the use of SA 479, UNS 21800 (Nitronic 60) to be acceptable for use in the Quickloc design, and is fabricated to ensure it is not sensitized. However, since the applicant stated that this material is not welded, the staff requested how the Quickloc mechanism is attached to the reactor vessel head.

In its response to RAI-SRP5.2.3-CIB1-03, Revision 0, dated January 15, 2010, and in response to RAI-SRP5.2.3-CIB1-03, Revision 1, dated March 21, 2010, the applicant stated that the Nitronic 60 material is used for the Quickloc plug and Quickloc nut, which are not welded, since these are mechanical parts. The Quickloc plug is inserted into the Quickloc instrument nozzle. The Quickloc instrument nozzles are welded to a weld-buildup of alloy steel on the reactor vessel head. The staff notes that this weld buildup (Material Specifications SFA 5.5, 5.23 or 5.28 of Section II to the ASME Code) is deposited on the top of the reactor vessel head and then machined to form a nozzle that penetrates the reactor vessel head. This weld build up nozzle has corrosion-resistant cladding applied to the inside diameter similar to the rest of the reactor vessel head. The applicant also stated that the buttering (Alloy 52/152) is applied to the top of the weld buildup to facilitate the welding of the Quickloc instrument nozzle to this buttering using Alloy 52/152. Therefore, there is a dissimilar metal weld using Alloy 52/152 attaching the

SA-182, Type 304 stainless steel Quickloc instrument nozzle to the Alloy 52 buttering on the alloy steel weld buildup nozzle. The applicant also provided information from the design drawing that the Quickloc instrument nozzle is welded to the Alloy 52/152 buttering after final post-weld heat treatment of the reactor vessel head. The staff finds that welding the Type 304 stainless steel Quickloc instrument nozzle after the final post-weld heat treatment of the reactor vessel closure head and Quickloc nozzle weld buildup minimizes the sensitization of the stainless steel Quickloc instrument nozzle. Therefore, the staff finds that the fabrication sequence of the Quickloc parts to the reactor vessel head using ASME Code, Section III ensures the integrity of the weld, including the final post-weld heat treatment of the reactor vessel head with the Quickloc nozzle weld buildup to ensure the material properties of the low alloy steel are retained.

In addition, the applicant stated that the Quickloc mechanism contains both pressure boundary and non-pressure boundary (internal structure) parts. The pressure boundary parts meet the requirements of ASME Code, Section III, Subsection NB, while the non-pressure boundary parts meet the requirements of ASME Code, Section III, Subsection NG. The staff finds the design and fabrication of the pressure boundary to be acceptable since it is evaluated in accordance with Class 1 requirements of ASME Code, Section III. The NRC staff also finds it appropriate for the design and fabrication of the non-pressure boundary parts to be designed and fabricated to the requirements of ASME Code, Section III, Subsection NG, which applies to reactor vessel internals and core support structures.

In regard to inspection during fabrication, the applicant provided the specific inspections to be performed for the pressure boundary parts in its response to RAI-SRP5.2.3-CIB1-03, Revision 0, dated January 15, 2010, and in response to RAI-SRP5.2.3-CIB1-03, Revision 1 dated April 21, 2010. These inspections included radiography and ultrasonic testing of the weld buttering and dissimilar metal weld. The weld buildup would also receive a magnetic particle inspection and an ultrasonic inspection, while the cladding and dissimilar metal weld would receive a liquid penetrant inspection. The inspections for the buttering and cladding would be performed in accordance with ASME Code, Section V, while the remainder of the inspections would be performed in accordance with Section III of the ASME Code, Subsection NB-5000. Since the fabrication inspections include volumetric examination of all the welds, and the inspections are in accordance with the ASME Code, the staff finds these inspections acceptable to ensure the integrity of the welds and base material.

However, to ensure the integrity of the welds is maintained, the NRC staff asked what type of inservice inspection is performed for these welds. In a letter dated January 15, 2010, the applicant stated that the dissimilar metal weld joining the stainless steel Quickloc instrument nozzle to the Alloy 52 buttering on the alloy steel Quickloc weld buildup requires an inservice inspection in accordance with Section XI of the ASME Code, Table IWB-2500-1, Item Number B5.10 (Category B-F). The NRC staff finds that Category B-F, Item Number B5.10 for dissimilar metal welds in pipe sizes larger than 10.2 cm (4 in) is the appropriate category for this weld to verify that the structural integrity is maintained as required by 10 CFR Part 50, Appendix A, GDC 30 and GDC 32, "Inspection of Reactor Coolant Pressure Boundary."

In its response to RAI-SRP5.2.3-CIB1-03, Revision 1, the applicant stated that the Quickloc nozzle weld buildup is a base metal weld buildup, and is considered an extension of the reactor vessel head forging and, therefore, would require no inservice inspections. The NRC staff does not agree that inservice inspections are not required, since even a non-structural attachment weld to the reactor vessel, Examination Category B-K in the ASME Code, Section XI, requires a surface examination for inservice inspections. The staff also notes that the Quickloc nozzle

weld buildup (maximum height of 37.03 cm (14.58 in) with a 19.1 cm (7.5 in) outside diameter and approximately 2.5 cm (1 in) thick) is designed as a nozzle penetration in the reactor vessel head, which forms the reactor coolant pressure boundary. Therefore, the overall dimensions of the weld buildup and the fact that this weld buildup serves as both a pressure boundary for the reactor coolant and a structural member to attach the Quickloc mechanism would necessitate that an inservice inspection be performed in order to verify that the structural integrity of the weld buildup is maintained as required by 10 CFR Part 50, Appendix A, GDC 30 and GDC 32.

In its response to RAI-SRP5.2.3-CIB1-03, Revision 3, dated August 3, 2010, the applicant stated that the inservice inspection of the Quickloc nozzle weld buildup is not a design issue and, therefore, should be addressed in the inservice inspection program to be developed by the COL applicant. Therefore, the applicant proposed a COL item in Section 5.3.6.6 of the AP1000 DCD, which states:

> The Combined License holder will establish an in-service inspection program prior to fuel load. The in-service inspection program will include the performance of a 100 percent volumetric examination of the weld build-up on the reactor vessel head for the instrumentation penetrations (Quickloc) conducted once during each 120-month inspection interval in accordance with the ASME Code, Section XI. The weld buildup shall meet the acceptance standards of ASME Code, Section XI, IWB-3514. Personnel performing examinations and the ultrasonic examination systems shall be qualified in accordance with ASME Code, Section XI, Appendix VIII. Alternatively, the Combined License holder may develop an alternative inspection in conjunction with the voluntary consensus standards bodies (i.e., ASME) and submit to the NRC for approval.

The staff agrees that the COL applicant should include the appropriate inservice inspection for the Quickloc nozzle weld buildup, since Section XI of the ASME Code does not specifically address this type of weld buildup on the reactor vessel head. The NRC also agrees that a volumetric examination (ultrasonic) of all eight Quickloc nozzle weld buildups performed with procedures and personnel with similar qualifications (ASME Code, Section XI, Appendix VIII) for other Class 1 ASME Code welds in the reactor coolant pressure boundary provides assurance of the integrity of this reactor coolant pressure boundary weld. The staff also finds the acceptance criteria of IWB-3514 for Category B-F welds in Section XI of the ASME Code is the appropriate acceptance standard for detecting the type of flaws that could affect the integrity of the Quickloc nozzle weld buildup and the reactor coolant pressure boundary. The staff notes that the inspection of the Quickloc nozzle weld buildup can be performed in conjunction with the adjacent dissimilar metal weld using the same inspection procedures. As experience is gained on this weld, technical basis for alternative examinations can be developed through the voluntary consensus standards body process, such as ASME, and submitted for NRC approval. Therefore, the staff finds that this COL item adequately addresses the concern for inservice inspection of the Quickloc nozzle weld buildup, because it ensures that the COL applicant's inservice inspection program will include the appropriate inservice inspection for this weld, thereby assuring its integrity as required by 10 CFR Part 50, Appendix A, GDC 30 and GDC 32. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

5.2.3.2.4 Conclusions

In summary, the staff finds that the requested modification for the materials is acceptable because it satisfies the requirements in Section VIII B.5.b or Section VIII B.5.c of

10 CFR Part 52, Appendix D, "Design Certification Rule for the AP1000 Design." The staff concludes that the changes to the pressure boundary materials in Revision 1 to TR-33 are technically acceptable. Section 4.5.1 of this report discusses the evaluation of the material changes for the CRDM components. The staff concludes that the TR-33, Revision 1, changes have been included in the AP1000 DCD are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.3 Reactor Vessel

5.3.2 Reactor Vessel Materials

5.3.2.1 Summary of Technical Information

In Revision 17 to the AP1000 DCD, the applicant proposed to resolve COL Information Item 5.3-3 (COL Action Item 5.3.2.2-1) by addressing the surveillance capsule lead factors and azimuthal locations in TR-23, "Surveillance Capsule Lead Factor and Azimuthal Location Confirmation," APP-GW-GLR-023, Revision 0, of September 2006. The applicant submitted TR-23 for staff review to demonstrate that it has met the requirements of COL Information Item 5.3-3. The submittal proposes a design change to locate the surveillance capsules in the minimum flux azimuthal locations to achieve lead factors in the range of 1.8 to 2.3, and as such satisfies the requirements in 10 CFR Part 50, Appendix H, "Reactor Vessel Surveillance Material Requirements," regarding surveillance capsule lead factors. The proposed change will eliminate the need for COL applicants to address the surveillance capsule lead factors and azimuthal locations requirement of COL Action Item 5.3.2.2-1.

In Section 5.3.6.3 of the AP1000 DCD, Revision 15, the COL action item states the following:

> The Combined License applicant will address confirmation of the surveillance capsule lead factors and azimuthal locations through an analysis which includes modeling of the capsule/holder.

In Revision 17 to the AP1000 DCD, the applicant proposed to resolve COL Information Item 5.3-3 by addressing the surveillance capsule lead factors and azimuthal locations in TR-23. The proposed revision to Section 5.3.6.3 of the DCD states the following:

> The Combined License information requested in this subsection has been completely addressed in APP-GW-GLR-023 (Reference 7), and the applicable changes are incorporated into the DCD. No additional work is required by the Combined License applicant to address the Combined License information requested in this subsection.
>
> The following words represent the original Combined License Information item commitment, which has been addressed as discussed above:
>
> > The Combined License applicant will address confirmation of the surveillance capsule lead factors and azimuthal locations through an analysis which includes modeling of the capsule/holder.

5.3.2.2 Evaluation

The surveillance capsule location in Figure 5.3-4 in Revision 15 of the DCD is at 29.70 degrees on either side of the 0- to 180-degree axis. The total number of surveillance capsules is eight. In this location, the estimated lead factor is outside (higher than) the range recommended in ASTM Standard E 185-82, "Standard Practice for Conducting Surveillance Test for Light-Water Cooled Nuclear Power Reactor Vessels," incorporated by reference into Appendix H to 10 CFR Part 50. To resolve this issue, the applicant proposed to use the locations (at the azimuthal minimums of the neutron flux, E > 1.0 million electron volts) at 45, 135, 225, and 315 degrees. However, the 45-degree location cannot be used because of mechanical interference with reactor internals; therefore, it is proposed to have three capsules per location at the 135- and 315-degree locations and two capsules at 225 degrees.

The applicant performed an analysis for the anticipated range of lead factors at the proposed locations, and the results indicate that the values are in the range 1.8 to 2.3 (i.e., within the required values in Appendix H). The calculations were carried out for an equilibrium core power distribution and adhere to the guidance in RG 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence." As stated in the guide, the cross-sections used (BUGLE-96) are based on the ENDF/B-VI file, the scattering cross-section approximation is the P_3 Legendre polynomial expansion, the angular quadrature approximation is S_8, and a 1/8 core symmetry was used. Because the 225-degree location contains only two capsules, two 1/8 core symmetry models were developed, one each for the two- and the three-capsule arrangements. Both models are based on the synthesis method; that is, the three-dimensional power distribution is derived from the synthesis of the (r, θ) and (r, z) distributions. The flux and fluence distributions derived in this manner are acceptable because the applicant followed the guidance in RG 1.190. This satisfies the requirements in GDC 14, "Reactor Coolant Pressure Boundary," GDC 30, and GDC 31.

Section 5.3.6.3 of the DCD, Revision 17, references TR-23 in its entirety. Although the entire report can be referenced for additional information, the staff finds that for clarity and completeness of the DCD as a stand-alone document, the DCD should include portions of the TR describing important design details. The staff issued RAI-SRP5.3.1-CIB1-01 on February 19, 2008, asking the applicant to include the following information in the AP1000 DCD: (1) the azimuthal locations of the capsules (in degrees) and the basis for these locations; (2) the calculated lead factors; and (3) Figure 1, "Surveillance Capsule Azimuthal Location." The applicant's RAI response, dated March 28, 2008, appropriately described the changes to the DCD to incorporate information from the TR. These design details were incorporated into Section 5.3.2.6 and Figure 5.3-4 of Revision 17 and subsequent DCD revisions.

In addition, the submittal states that the applicant examined the presence of the surveillance capsules in the proposed locations and concluded that there is no interference with required potential actions to mitigate severe accidents.

The staff found that the DCD, Revision 16 changes discussed by the applicant in TR-23 meet the requirements of Appendix H to 10 CFR Part 50 and are acceptable. Therefore, AP1000 COL Information Item 5.3-3 is resolved. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the changes incorporated into Revision 17 contribute to the increased standardization of the certification information in the AP1000 DCD.

5.3.2.3 Conclusions

On the basis of its review of TR-23 and the associated changes to the DCD, the staff concludes that the proposed change meets the requirements of Appendix H to 10 CFR Part 50, and that the applicant has provided sufficient design information to close out COL Information Item 5.3-3. Furthermore, the staff finds that the TR-23 conclusions regarding the surveillance capsule lead factors and azimuthal locations are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.3.3 Pressure-Temperature Limits

5.3.3.1 Introduction

In Revision 16 of the AP1000 DCD, the applicant included generic bounding P/T limits. In a letter dated May 30, 2008, the applicant submitted APP-RXS-Z0R-001, "AP1000 Generic Pressure Temperature Limits Report," Revision 1, which describes the generic pressure-temperature limits report (PTLR) for the AP1000. The generic bounding P/T limits in Revision 17 to the DCD are the same P/T limits used in the generic PTLR. The applicant plans to have AP1000 COL applicants use the generic PTLR described in APP-RXS-Z0R-001, Revision 1, when developing their plant-specific P/T limits.

The applicant submitted TR-33 and supporting information in a letter dated June 7, 2007, to provide technical justification for the proposed changes. The staff's evaluation of the proposed changes is provided below.

5.3.3.2 Evaluation

The NRC has established requirements in 10 CFR Part 50 to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates P/T limit curves based on the following NRC regulations and guidance: 10 CFR Part 50, Appendix G; RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," issued May 1988; and NUREG-0800 Section 5.3.2. RG 1.99 describes the methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron irradiation. Appendix G to 10 CFR Part 50 requires that P/T limit curves for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the methodology of Appendix G to ASME Code, Section XI.

a. Background

NUREG-0800 Section 5.3.2 provides an acceptable method of determining the P/T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to ASME Code, Section XI. The basic parameter of this methodology is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. Appendix G to ASME Code, Section XI, requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 on stress intensities resulting from hydrostatic testing. Appendix G to ASME Code, Section XI, also requires a safety factor of 1.0 on stress intensities resulting from thermal loads for normal and transient operating conditions as well as for hydrostatic testing. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is

normal to the direction of the maximum stress (i.e., of axial orientation). This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to six times its depth. The critical locations in the RPV beltline region for calculating heatup and cooldown P/T limit curves are the 1/4 thickness (1/4T) and 3/4T locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively. The methodology found in Appendix G to ASME Code, Section XI, requires that licensees determine the ART (or adjusted RT_{NDT}) at the 1/4T and 3/4T locations. The ART is defined as the sum, the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin term.

RG 1.99 provides guidance on the determination of ΔRT_{NDT} and the margin term. ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence, and the calculation procedures.

b. The Applicant's Evaluation

In RAI-TR33-001, dated April 23, 2007, the staff asked the applicant to provide the following information:

1. Detailed methodology used in the development of the P-T limit curves.
2. Beltline material properties (Cu, Ni) assumed, including initial RT_{NDT} of materials.
3. Fluence assumed in the calculation of adjusted of RT_{NDT}.
4. P/T data points.

The applicant provided this information in a letter dated June 7, 2007, since it plans for the curves to be used by the COL holders as bounding curves.

The methodologies provided by the applicant are consistent with those in Westinghouse Commercial Automatic Power (WCAP)-14040-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," Revision 4 of May 2004. The P/T limits, which are valid for up to 54 effective full-power years (EFPYs) of operation, were calculated with chemistry factors obtained from the tables in RG 1.99. As discussed in Section 5.2.3.2.2 of this report, the applicant proposed (and the staff agrees with) changing the maximum wt% of copper to 0.06 percent. This is the value the applicant used for the calculations of AP1000 bounding P/T limit curves. The initial RT_{NDT} values are generic values for the design. Table 5-2 shows the values the applicant has provided for limiting ART calculations of the beltline girth welds at two locations.

Table 5-2. Values for Limiting ART Calculations of the Beltline Girth Welds

	1/4T Location	3/4T Location
Fluence at 54 EFPYs	1.510×10^{19}	5.51×10^{18}
Chemistry Factor	82	82
ΔRT_{NDT}	32.9 °C (91.3 °F)	20.2 °C (68.3 °F)

Table 5-2. Values for Limiting ART Calculations of the Beltline Girth Welds

	1/4T Location	3/4T Location
Initial RT_{NDT}	-28.9 °C (-20 °F)	-28.9 °C (-20 °F)
Margin	18.6 °C (65.5 °F)	18.6 °C (65.5 °F)
ART	58.3 °C (137 °F)	45.6 °C (114 °F)

In addition, the applicant provided P/T data points without margins for instrumentation errors. The staff used this information to evaluate the acceptability of the proposed bounding P/T limit curves.

c. Evaluation

As discussed previously, the applicant requested approval for bounding P/T limit curves to be used by COL holders with the AP1000 design. The staff performed independent calculations of the ART values using the methodology in RG 1.99. Based on these calculations, the staff verified the limiting ART values at the beltline girth welds.

Given the acceptability of the applicant's calculated ART values for the limiting beltline material to 54 EFPYs, the staff evaluated the P/T limit curves for acceptability by performing a finite set of check calculations based on information submitted by the applicant and by using the methodologies referenced in the ASME Code (as indicated in NUREG-0800 Section 5.3.2). The staff verified that the proposed P/T limit curves satisfy the requirements in Section IV.A.2 of Appendix G to 10 CFR Part 50.

In addition, Appendix G to 10 CFR Part 50 also imposes a minimum temperature at the closure flange region based on the reference temperature for the flange material. Section IV.A.2 of Appendix G to 10 CFR Part 50 states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature at the closure flange region, which is highly stressed by the bolt preload, must exceed the reference temperature of the material in that region by at least 71.1 °C (160 °F) for core critical operation, 48.9 °C (120 °F) for normal, noncritical core operation, and by 32.2 °C (90 °F) for hydrostatic pressure tests and leak tests. Based on this limiting flange reference temperature, the staff has determined that the proposed P/T limits have satisfied the above requirements for the closure flange region during all modes of normal operation and for hydrostatic pressure and leak testing.

Based on this independent assessment, the staff concludes that the applicant's proposed P/T limit curves meet the requirements of Appendix G to 10 CFR Part 50 and are acceptable for operation of an AP1000 design through 54 EFPYs of facility operation. This determination; however, is only valid for the material properties and the projected fluence identified in this SER. Any changes to these values will require additional review. Revision 17 to the AP1000 DCD includes these revised P/T limit curves.

These revised P/T limits are generic and may be used by COL holders referencing the AP1000 certified design, contingent upon verification by the COL holder of the material properties and fluence projection. At this time, the NRC has not issued a COL for any AP1000 plant. Therefore, the proposed P/T limits contribute to the increased standardization of the certification information in the AP1000 DCD.

In AP1000 DCD, Revision 15, COL Information Item 5.3.6.1, the applicant stated that COL applicants will address the use of plant-specific P/T limits. In TR-6, "AP1000 As-Built COL Information Items," APP-GW-GLR-021, Revision 0 of June 2006, the applicant proposed to change the responsibility from the COL applicant to the COL holder. However, in subsequent discussions between the NRC staff and the applicant, the applicant decided to provide a generic PTLR in conjunction with a future AP1000 DCD amendment for use by AP1000 COL applicants. Upon review and approval of the PTLR, COL applicants will be able to use it for their respective plants as long as the PTLR methodology remains the same. In conjunction with the Bellefonte reference COL application review, the staff is requesting the COL applicant provide license conditions in which the COL holder will be required to: (1) update its P/T limits using the PTLR methodologies approved in the AP1000 DCD and using plant-specific material properties; and (2) to inform the NRC of its plans to use updated P/T limits. The applicant provided its generic PTLR for NRC review and approval in a letter dated May 30, 2008. The NRC discussed its evaluation of the PTLR in a letter to the applicant dated December 30, 2008. However, the staff finds that the applicant needs to provide the PTLR reference in its DCD. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In reviewing the AP1000 PTLR, the staff noted one inconsistency between the AP1000 PTLR and Revision 17 to the DCD concerning the listed nickel content of the circumferential reactor vessel beltline weld. Table 3 of the AP1000 PTLR listed the maximum allowable nickel content as 0.95 percent by weight. However, Table 5.3-1 in Revision 17 of the DCD listed the maximum allowable nickel content as 0.85 percent. In RAI-SRP5.3.2-CIB1-01 dated July 31, 2008, the NRC asked the applicant to resolve this discrepancy by amending either the DCD or the AP1000 PTLR to specify the same maximum nickel content of 0.85 percent or 0.95 percent. In its response dated September 5, 2008, the applicant indicated that the actual limiting value of the nickel content of the reactor vessel beltline weld is 0.85 percent by weight. Furthermore, the applicant indicated that this limiting nickel content value would be changed in Table 3 of the AP1000 PTLR to be 0.85 percent, consistent with the value established in Section 5.3 of the AP1000 DCD, Revision 17. In a letter dated April 2, 2009, the applicant provided Revision 2 of the AP1000 PTLR, which includes the proposed changes. The staff finds that the applicant has appropriately addressed the RAI and has verified that Revision 2 of the AP1000 PTLR incorporates the proposed revision to the maximum nickel content. As a result, RAI-SRP5.3.2-CIB1-01 is closed.

Criterion 3 of PTLR (Reference generic letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits") requires the evaluation of the LTOP system. AP1000 Technical Specification Limited Condition for Operation (LCO) 3.4.14 does not specify the normal RNS suction relief valve lift pressure setpoint. The applicant should revise LCO 3.4.14 to state that the RNS suction relief valve lift setpoint be within the specified limit in the PTLR. In a letter dated May 20, 2009, the applicant provided a supplemental response to RAI-SRP5.3.2-CIB1-02, which includes a revision to the PTLR with the following addition:

2.1 Low Temperature Overpressure Protection (LTOP) System

> The Normal Residual Heat Removal System (RNS) pump suction line relief valve will have a lift setpoint of 500 psig with a full open pressure of 550 psig. The lift setpoint has been developed using the NRC approved methodology specified in Reference 2.

From the review of the applicant's response, the staff concludes that the provisions in the response are acceptable because they adequately address PTLR Criterion 3 of GL 96-03. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

5.3.3.3 Conclusions

The staff finds that the proposed bounding P/T limit curves meet the requirements of Appendix G to 10 CFR Part 50 and are acceptable for the stated material properties and projected fluence for 54 EFPYs. Furthermore, the staff finds that the TR-33 conclusions regarding the bounding P/T limit curves are generic and are expected to apply to all COL applications referencing the AP1000 DC. The staff reviewed and approved the applicant's PTLR methodology for the AP1000. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.3.6 Reactor Vessel Insulation

The applicant has completed the design of the reactor vessel insulation system (RVIS). In TR-24, "Reactor Vessel Insulation System - Verification of In-Vessel Retention Design Bases," APP-GW-GLR-060, the applicant provided information to demonstrate that the RVIS is designed to provide adequate cooling to ensure in-vessel retention of a damaged and relocated core. On this basis, the applicant proposed to close COL Information Item 5.3-5 (COL Action Item 19.2.3.3.1.3.2-1). The staff's evaluation of the RVIS design is in Section 19.2.3.3.1.3.2 of this report.

5.4 Reactor Coolant System Component and Subsystem Design

5.4.1 Reactor Coolant Pump Assembly

5.4.1.1 Summary of Technical Information

In Section 5.4.1 of Revision 17 of the DCD, the applicant proposed to make changes related to the certified RCP design. These changes include: (1) change of the canned motor RCP design description to a more generic sealless pump; (2) use of an externally mounted heat exchanger; (3) change of the RCP flywheel design from depleted uranium to bimetallic construction; and (4) other miscellaneous changes.

5.4.1.2 Pump and Motor Design

The certified AP1000 RCP design, as described in Section 5.4.1.2.1, "Design Description," of Revision 15 to the DCD, is a single-stage, hermetically sealed, high-inertia, centrifugal canned motor pump. In TR-34, "AP1000 Licensing Design Change Document for Generic Reactor Coolant Pump," submitted by letter dated November 17, 2006, the applicant proposed to replace the specific term "canned motor pump" with a generic sealless pump design to provide flexibility in the selection of a specific pump design and pump vendors. In DCD Revision 17, the applicant revised Section 5.4.1.2.1 to replace the term "canned-motor pump" with the phrase "sealless pump of either canned motor or wet winding design." In addition, in many DCD Tier 2 sections (e.g., Sections 1.9.3, 1.9.4.2.3, 1.9.5.1.6, 3.5.1.2.1.4, 3.9.2.3, 4.4.4.6, 5.1.2, and 5.1.3.3 and Table 5.1-2) associated with the RCPs, the applicant replaced the term "canned-motor pump" with "sealless pump," or "sealless pump of canned motor design." In

DCD Tier 1, Section 2.1.2, "Reactor Coolant System," the applicant replaced the term "canned motor reactor coolant pumps" in the design description with the phrase "sealless reactor coolant pumps."

The canned motor pump design contains the motor and all rotating components inside a pressure vessel designed for full RCS pressure. Since the shaft for the impeller and rotor is contained within the pressure boundary, seals are not required to restrict leakage out of the pump into containment. Thus, the concern regarding RCP seal failure is eliminated. The staff agrees that the use of the generic term "sealless pump" achieves the same objective as specifying a "canned motor pump" with regard to the seal failure concern. However, the NRC had based its review and acceptance of the AP1000 DCD on the canned motor pump design. In DCD Revision 16, the applicant did not provide materials related to the wet winding pump design.

Revised DCD Figure 5.4-1, "Reactor Coolant Pump," and Reference 10 in DCD Section 5.4.16 are based on a canned motor pump. In the June 7, 2007, response to RAI-TR34-1, the applicant stated that the RCP design and licensing basis for the standard AP1000 plant is the canned motor RCP. In the October 5, 2007, response to RAI-TR34-SRSB-01, the applicant stated that for an applicant to use the wet winding sealless RCP, Reference 10 in Section 5.4.16 and Figure 5.4-1 of the DCD would need to be changed in the COL FSAR to incorporate the wet winding sealless design. This change would be a departure from the DCD and would open licensing review of the design change. Detailed design information on a wet winding sealless RCP design would be needed for NRC review and approval. In the April 11, 2008, response to RAI-SRP5.4.1-CIB-01, the applicant proposed changes to DCD Revision 16 to identify the canned motor design as the RCP design for the standard design, eliminating the mention of the wet winding motor design from the DCD. In DCD Revision 17, the applicant revised Section 5.4.1.2.1 to state that the RCP is a "single-stage, hermetically sealed, high-inertia, centrifugal sealless pump of canned motor design." This sentence is re-classified as Tier 2* information in that staff approval is required before implementing a change in this information. Therefore, the AP1000 design basis for the RCP is a sealless pump of canned motor design. Use of a sealless pump design of another type, such as a wet winding motor design, would be a departure from the DCD and would require detailed design information in the COL FSAR for NRC review and approval. The staff finds this acceptable.

In DCD Tier 1, Section 2.1.2, "Reactor Coolant System," the applicant also replaced the term "canned motor reactor coolant pumps" in the design description with the phrase "sealless reactor coolant pumps."

5.4.1.3 Heat Exchanger Design

In DCD Revision 17, the applicant proposed to change the heat removal design of the RCP by using an externally mounted, conventional shell and tube heat exchanger and a stator cooling jacket to replace the existing thermal barrier internal cooling coils and wraparound heat exchanger configuration. The applicant revised DCD Sections 5.4.1.2.1 and 5.4.1.2.2 to describe the revised motor cooling arrangement. An auxiliary impeller at the lower part of the rotor shaft circulates a controlled volume of the reactor coolant through the motor cavity, where the rotor, bearing and stator are cooled, and through an external heat exchanger where the coolant is cooled to about 65 °C by the component cooling water (CCW) circulating on the shell side. The CCW also circulates through a cooling jacket on the outside on the motor housing to cool the stator. The applicant revised DCD Figure 5.4-1 to show the external heat exchanger configuration. In addition, it revised DCD Section 5.4.1.3.3, "Pressure Boundary Integrity," to

include the external piping and tube side of the external heat exchangers as a part of the pressure boundary components that meet the requirements of ASME Code, Section III.

In TR-34, the applicant explained that it changed to an external heat exchanger for the RCP because as the detailed design of the pump progressed, the increased heat transfer requirements on the heat exchanger resulting from increased motor power requirements and the effects of design transients on motor operation have resulted in significant manufacturing challenges associated with the wraparound heat exchanger design. Therefore, a conventional shell and tube heat exchanger mounted on the pump flange is implemented to replace the current wraparound heat exchanger.

In its June 7, 2007, response to RAI-TR34-2, the applicant summarized the RCP cooling design. This includes the heat source from motor electrical loss, fluid and friction losses, hot primary coolant crossing the thermal barrier into the motor, and the heat removal capacity of heat exchanger and water jacket. The external heat exchanger is specified to remove 2.4 megawatts with 540 gallons per minute (gpm) of CCW at 35 °C and 600 gpm of primary flow at 69 °C. In the October 5, 2007, supplemental response to RAI-TR34-SRSB-02, the applicant indicated that the RCP design specification defining the external heat exchanger design requirements is passed to the external heat exchanger supplier.

Establishment of the design requirements allows for the finalization of the design. The external heat exchanger generic design report will provide detailed design information for the external heat exchanger. In its July 3, 2008, response to RAI-SRP5.4.1-SRSB-01, the applicant indicated that this design report would be available for NRC review on October 31, 2008. In addition, each pump would be performance tested with the heat exchanger intended for field use before shipment. The applicant also indicated that the auxiliary impeller has been designed for an Euler head rise of 240 feet (ft) at 1782 revolutions per minute. Prototype RCP testing in the future will verify that the actual bearing water flow rate is sufficient to satisfy design requirements. If an unpredicted difference occurs between the calculated bearing water flow rate and the measured test value, an easily implemented design change of adding an annular ring to the motor shaft may be executed to increase auxiliary impeller flow capacity.

The staff identified review and acceptance of the external heat exchanger design report as Open Item OI-SRP5.4.1-SRSB-01.

In a letter dated March 26, 2010, the applicant submitted a response to Open Item OI-SRP5.4.1-SRSB-01 that included the external heat exchanger design specifications regarding its functions and design requirements. The design specifications specify the objective of the external heat exchanger to cool primary side water used in a RCP motor, and specify that the primary-side and secondary-side components of the heat exchanger shall meet Section III, Division 1, Class 1 and Class 3, respectively, of the ASME Code. In addition, the external heat exchanger shell will serve as supports for Section III, Class 1 heat exchanger tubes that form part of the primary pressure boundary, and as such cannot be allowed to fail in such a way as to prevent the Class 1 pressure boundary from performing its function. The heat exchanger design requirements specify the heat exchanger configuration and nozzle size, the design basis life of 60 years, normal operating conditions, allowable pressure drop of the primary and secondary sides, design pressure and temperature, and thermal transient operating conditions, material, and vibration analysis. Regarding the vibration analysis, it specifies that the heat exchanger shall be designed to avoid flow-induced tube vibration under maximum expected flows and design temperature conditions. The adequacy with respect to flow-induced vibration shall be demonstrated by either: (a) an identical heat exchanger having operated satisfactorily

under flow and temperature conditions at least as severe as those expected for the heat exchanger; or (b) the maximum expected secondary side flow velocities will be below the critical velocities for fluid-elastic excitation, and the lowest tube natural frequency exceeds by a factor of 1.5 the vortex shedding frequency calculated for the maximum expected secondary side flow velocity. Based on this review, the staff concludes that the applicant has successfully addressed the open item; therefore, Open Item OI-SRP5.4.1-SRSB-01 is closed.

5.4.1.4 Reactor Coolant Pump Flywheel Integrity

5.4.1.4.1 Summary of Technical Information

In Revision 16 to the AP1000 DCD, the applicant proposed to change the RCP design from a canned motor RCP with a depleted uranium flywheel to a generic sealless RCP with a bimetallic flywheel assembly. TR-34 identified changes to the design of the RCP and the technical justification for the proposed changes.

The AP1000 DCD, Revision 15, had specified a canned motor RCP with a depleted uranium flywheel. The Revision 16 changes to the AP1000 DCD specify a generic sealless RCP with a bimetallic flywheel assembly. Changing to a generic sealless RCP will provide flexibility in selecting a specific pump design and thus increase the number of possible pump vendors. Revision 16 to the AP1000 DCD includes a revised flywheel assembly design of bimetallic construction.

Concerning the material specifications for the flywheel, TR-34 stated that the preliminary flywheel design (in Revision 15 of the AP1000 DCD) employed an upper and lower flywheel assembly constructed of forged depleted uranium disks fitted to an inner stainless steel hub, which was fitted to the motor shaft. Structural integrity of the flywheel assemblies relied upon the strength of the depleted uranium forged disks. The depleted uranium flywheel was designed to meet the minimum rotating inertia value, 695.3 kilograms - square meter ($kg\text{-}m^2$) (16,500 pounds - square foot ($lb\text{-}ft^2$)), given in DCD Tier 2, Table 5.4-1. As the design progressed, it was determined that this inertia value needed to be increased to meet the pump coastdown used in the safety analyses as given in DCD Figure 15.3.2-1. To achieve the required inertia, the depleted uranium flywheel design required increases in length and/or diameter. However, increases in diameter resulted in stress levels beyond the design criteria limits and increases in length resulted in violation of the RCP space envelope as well as unacceptable rotor dynamics. Therefore, a revised flywheel assembly design of bi-metallic design with heavy tungsten alloy inserts was developed.

In a letter dated September 28, 2007, the applicant provided additional information concerning the materials and design of the flywheel assembly. The design features heavy tungsten alloy annular or cylindrical segments, which are machined and fitted around a central Type 403 stainless steel hub. The segments are held in place by an interference fit of an 18-Ni maraging steel retainer cylinder placed over the outside of the assembly. Alloy 690 endplates and an outer thin shell hermetically seal the assembly from primary coolant. Structural integrity of the flywheel assembly relies upon the stainless steel hub and the retainer cylinder. Both the upper and lower flywheels are of the same design.

In a letter dated June 12, 2007, the applicant submitted TR-106, “AP1000 Licensing Design Changes for Mechanical Systems and Component Design Updates,” which revised the flywheel enclosure (endplates and outer thin shell) materials. Design change 245 in TR-106 stated that the applicant would revise Section 5.4.1.3.6.3 of the AP1000 DCD by replacing references to

"Alloy 690" with the phrase "corrosion resistant material" for the flywheel enclosure (endplates and outer thin shell) to meet the intent of creating a generic RCP design description that would facilitate other future RCP suppliers. In addition, the applicant stated that the RCP supplier had proposed the material change from Alloy 690 to Alloy 625 because of the low coefficient of thermal expansion and better weldability of Alloy 625. In a letter dated October 5, 2007, the applicant provided supporting information concerning the test results and experience of using Alloy 625 in contact with the reactor coolant. The applicant also concluded that the AP1000 DCD should not include such detailed design information as referencing the specific material (i.e., Alloy 625); therefore, the AP1000 DCD would only specify "corrosion-resistant material."

5.4.1.4.2 Evaluation

GDC 4 requires that SSCs important to safety shall be appropriately protected against environmental and dynamic effects, including the effects of missiles that may result from equipment failure. GDC 1 requires that SSCs important to safety shall be designed, fabricated, erected, and tested to quality standards to ensure a quality product in keeping with the required safety function. The staff's view of the Revision 16 changes related to Section 5.4.1.3.6.3 of the DCD is that they minimalize the potential for RCP flywheel failures and that the materials are adequate to ensure a quality product commensurate with the importance to safety.

Section 5.4.1.2.1 of the AP1000 DCD, Revision 16, changes the pump design from standard (canned motor) to a more generic "sealless" pump design (canned motor, wet-winding, etc.). The applicant revised Section 5.4.1.3.6.3 of the AP1000 DCD, Revision 16, to state that the analysis to determine the capacity of the housing to contain the fragments of the bi-metallic flywheel appears in Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, "Structural Analysis Summary for the AP1000 Reactor Coolant Pump High Inertia Flywheel," dated October 2006. This Curtiss-Wright report is only applicable for the canned-motor design. In RAI-SRP5.4.1-CIB1-01, the staff asked the applicant to change the DCD to provide the following:

- A specific RCP/flywheel design (i.e., a single-stage, high-inertia, centrifugal, sealless RCP of canned-motor design). Currently, this is the only RCP design that has a supporting analysis for the flywheel integrity and missile generation.

- The material specifications for the flywheel and the specific inspections to be performed on the flywheel.

- The material type for the end plates and outer shell since TR-106 replaced the-material type "Alloy 690" with "corrosion resistant" from AP1000 DCD, Revision 16, Section 5.4.1.3.6.3.

In a letter dated April 11, 2008, the applicant responded to RAI-SRP5.4.1-CIB1-01 and provided the following:

- RCP design and flywheel structural analysis: AP1000 DCD, Revision 16 will be changed to incorporate the standard design RCP (a single-stage, hermetically sealed, high-inertia, centrifugal, sealless pump of canned-motor design).

- Flywheel material and inspections: information identifying the materials used in the flywheel, and the inspections that will be performed on the flywheel.

- Flywheel enclosure material: information identifying the materials used in the flywheel enclosure.

The staff reviewed this information and provides the following summary and evaluation of each issue.

Flywheel Material and Inspections

The Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, dated October 2006, used the material properties from the material specifications for the flywheel (Type 403 stainless steel inner hub—ASTM A336, Grade F6, 18-Ni maraging steel outer hub—AMS 6519, Vascomax T250 (UNS 6075) and heavy tungsten inserts—ASTM B777, Class 4) to evaluate the flywheel integrity and ability to minimize missile generations. AP1000 DCD Section 5.4.1.3.6.3 also relies on the material specifications and preservice nondestructive testing to demonstrate the integrity of the flywheel and justify the removal of the flywheel from an inservice inspection program. Therefore, to ensure structural integrity of the flywheel, as evaluated in the Curtiss-Wright report, the applicant should change the AP1000 DCD to include these material specifications or reference the Curtiss-Wright report for the material specifications to be used. The staff previously addressed this in RAI-SRP5.4.1-CIB1-01. The staff identified this as Open Item OI-SRP5.4.1-CIB1-01.

In a letter dated May 26, 2009, in response to RAI-SRP5.4.1-CIB1-01, Revision 1, the applicant stated that the flywheel outer hub material was changed from 18-Ni maraging steel to 18Mn-18Cr alloy steel (ASTM A289, Grade 8) as a result of lessons learned during a flywheel mockup assembly, and that the 18Mn-18Cr alloy steel is not susceptible to stress corrosion cracking and hydrogen embrittlement. A proposed change to Section 5.4.1.3.6.3 of the AP1000 DCD to include the new material specification was also provided in this letter. The staff confirmed that this material is more resistant to stress corrosion cracking than 18-Ni maraging steel based on the current operating experience of 18Mn-18Cr alloy steel retaining rings on generators used since the mid 1980's. The generator environment is more aggressive due to the hydrogen cooling and the wet oxygenated environment compared to the PWR reactor coolant water, which controls the oxygen content. Also, the staff notes that this alloy steel has a high chromium content, which is similar to stainless steels currently used in the reactor coolant. The 18Mn-18Cr material is similar to a high strength stainless steel with yield strength of 195 kips per square inch (ksi) compared to 250 ksi for the 18-Ni maraging steel. Since this alloy steel is not a nickel based alloy, such as Alloy 600, primary water stress corrosion cracking is not a concern. The NRC also notes that the 18Mn-18Cr alloy steel outer hub will be enclosed in an Alloy 625 flywheel enclosure to prevent the outer hub from contacting the reactor coolant. Therefore, the staff finds the use of the 18Mn-18Cr alloy steel acceptable based on the current operating experience of this material in an aggressive stress corrosion environment, and even if the flywheel ruptures, the flywheel would be contained in the RCP as discussed below, thereby preventing a missile that could impact safety-related equipment or structures, which meets the requirements of GDC 4.

In a letter dated August 4, 2009, in response to RAI-SRP5.4.1-CIB1-01, Revision 2, the applicant provided the material specifications for the inner hub (ASTM A336, Grade F6) and the tungsten inserts (ASTM B777, Class 4) in a proposed change to Section 5.4.1.3.6.3 of the AP1000 DCD as requested by the staff, which partially resolves Open

Item OI-SRP5.4.1-CIB1-01. In a subsequent revision to the AP1000 DCD the material specifications as stated in letters dated May 26, 2009 and August 4, 2009 were included, which resolves this issue.

However, since the outer hub material was changed to 18MNn-18Cr, the evaluation of the flywheel in Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, dated October 2006, is no longer valid since it does not bound the new material (18Mn-18Cr). In a letter dated August 4, 2009, the applicant submitted a revised flywheel analysis, Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, Revision 2, dated July 2009, which included the new material (18Mn-18Cr), and is, therefore, part of Open Item OI-SRP5.4.1-CIB1-01, which is discussed in the "RCP design and flywheel structural analysis" section below.

In regard to the preservice inspection of the flywheel, the applicant provided changes to AP1000 DCD, Revision 17, Section 5.4.1.3.6.3, to include surface examinations and volumetric inspections of the inner hub and retainer cylinder ring, and an overspeed spin test followed by a visual inspection and leak test of the final assembly in accordance with ASME Code, Section III. In addition, impact/fracture toughness testing will also be conducted on the inner hub and retainer cylinder ring. The staff finds that these inspections provide reasonable assurance of the integrity of the flywheel during fabrication and ensure that the basis for safe operation of the RCP will be maintained.

Flywheel Enclosure Material

Design change 245 in TR-106 stated that the applicant would revise Section 5.4.1.3.6.3 of the AP1000 DCD by replacing references to "Alloy 690" with the phrase "corrosion resistant material" for the flywheel enclosure (endplates and outer thin shell) to meet the intent of creating a generic RCP design description. The NRC staff notes that Section 5.4.1.3.6.3 of the AP1000 DCD credits the use of the flywheel enclosure to prevent contact with the reactor coolant and to minimize the potential for corrosion of the flywheel and contamination of the reactor coolant. A leak in the flywheel enclosure during operation could result in an out-of-balance flywheel assembly. In addition, the applicant stated that the use of the required material specifications and nondestructive testing during fabrication of each flywheel demonstrates the quality of the flywheel, thereby permitting the removal of the requirement for periodic inservice inspections of the flywheel to ensure that the basis for safe operation of the RCP is maintained. Based on the above, the NRC staff notes that, although the flywheel enclosure is not credited for retaining potential flywheel missiles, the flywheel enclosure is an integral part of ensuring the integrity of the flywheel so that it will not generate a missile. In the course of its review, the staff identified that Section 5.4.1.3.6.3 of the AP1000 DCD should be revised to state the material type (i.e., Alloy 690 and/or Alloy 625) for the flywheel enclosure.

In a letter dated April 11, 2008, the applicant included Alloy 625 in Section 5.4.1.3.6.3 of the AP1000 DCD as the material used for the flywheel enclosure. Therefore, the NRC staff considers Alloy 625, as referenced in ASTM B-443, "Standard Specification for Nickel-Chromium-Molybdenum-Columbium Alloy (UNS N06625) and Nickel-Chromium-Molybdenum-Silicon Alloy (UNS N06219) Plate, Sheet, and Strip," and ASTM B-564, "Standard Specification for Nickel Alloy Forging," to be an acceptable material for the flywheel enclosure, based on current operating experience in fuel assemblies and testing performed by Bettis Atomic Power Laboratory, which was discussed in the applicant's letter dated October 5, 2007. With the inclusion of the material specification for the flywheel enclosure, the staff finds the change to Section 5.4.1.3.6.3 acceptable.

RCP Design and Flywheel Structural Analysis

Concerning the RCP design, see Section 5.4.1.2 of this report for the staff's evaluation. The staff concludes this standard design type (a single-stage, hermetically sealed, high-inertia, centrifugal, sealless pump of canned-motor design) is supported by an applicable flywheel integrity analysis, and it standardizes the certification information for all COL applicants.

The structural analysis of the bimetallic flywheel with heavy tungsten alloy inserts for the sealless RCP with canned motor and including a missile containment evaluation of a fractured flywheel was initially documented in Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, issued October 2006.

However, due to a change in material (18Mn-18Cr) for the outer hub of the flywheel, the applicant submitted a revised flywheel analysis, Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, Revision 2, dated July 2009, in a letter dated August 4, 2009. The August 4, 2009, letter also provided a proposed change to Section 5.4.16 of the AP1000 DCD, which references this new Curtiss-Wright analysis. This analysis used the material properties of 18Mn-18Cr, using a similar methodology as the previous analysis. However, the NRC staff noted that there were other changes to the RCP including material changes to the pump, dimensional changes to the flywheel, and an additional pump part (stator closure ring) adjacent to the upper flywheel that would contain the upper flywheel if it were to rupture. In addition, a missile generated from a fractured portion of the upper flywheel would now have to penetrate three parts (stator closure, stator closure ring, and thermal barrier) which are bolted together. Therefore, the staff identified this as part of Open Item OI-SRP5.4.1-CIB1-01, Revision 1, in that the analysis should also account for shearing of the bolts connecting these three parts.

In response to Open Item OI-SRP5.4.1-CIB1-01, Revision 1, in a letter dated June 6, 2011, the applicant stated that the new analysis assumes that the flywheel is contained by the three parts (stator closure, stator closure ring, and thermal barrier) which are bolted together, and act as one unit. These three parts act as one unit due to the combination of large compressive stresses from the main flange bolts and the recessed fits between the three parts, which transfer the shear load from one part to another. The NRC staff notes that the analysis is conservative since only a portion of the entire volume of the three parts (stator closure, stator closure ring, and thermal barrier) was credited in the analysis with respect to containing a fractured flywheel. Therefore, the NRC staff finds that the shear mechanism was analyzed and bounded by the analysis. In addition, Revision 2 of the analysis accounted for the dimensional changes in the flywheel, and the outer hub material properties of 18Mn-18Cr.

In response to RAI-SRP5.2.3-01, Revisions 1 and 2, dated December 31, 2009, and April 7, 2010, respectively, the applicant provided information on material changes to carbon steel for specific parts of the pump, which are the stator main flange, stator shell, and external heat exchanger supports. These parts are not exposed to reactor coolant and are not credited for containing a fractured flywheel. Therefore, the NRC staff agrees that these material changes to the stator main flange, stator shell, and external heat exchanger supports have no effect on the flywheel analysis. In addition, the staff finds the use of carbon steel for these parts acceptable, since they are not exposed to the reactor coolant. Therefore, the staff finds Open Item OI-SRP5.4.1-CIB1-01 completely resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The flywheel analysis, Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009-P, Revision 2, issued July 2009, demonstrates that the calculated stresses during both normal operating conditions and design conditions are less than the applicable stress limits. In addition, missile penetration calculations show that in the unlikely event of a flywheel fracture, the flywheel assembly components will not have sufficient energy to penetrate the pump pressure boundary structures. Therefore, the NRC staff finds the structural analysis for the bi-metallic flywheel with heavy tungsten alloy inserts acceptable because the missile penetration calculations show that if the flywheel fractures, the flywheel assembly components will not have sufficient energy to penetrate the pump pressure boundary structures; therefore, the design meets the requirements of GDC 4. The NRC staff had approved the previous AP1000 canned motor design with a depleted uranium flywheel assembly, which also relies on the pump casing to confine the flywheel if the flywheel fractures. The missile penetration calculations for both material designs used similar methodologies.

5.4.1.4.3 Conclusion

Based on the above evaluation, the staff finds that the material specifications used and the preservice inspections provide reasonable assurance of the flywheel integrity, and that the Curtiss-Wright Electro-Mechanical Corporation Report AP1000 RCP-06-009 demonstrates that if the flywheel assembly fails, the flywheel components will not penetrate the pump pressure boundary structures. Therefore, the staff finds the RCP flywheels acceptable since they meet the requirements of GDC 1, GDC 4, and ASME Code, Section III.

The NRC staff reviewed the changes in Section 5.4.1.3.6.3 of the AP1000 DCD and finds that the AP1000 DCD adequately incorporates the proposed changes, as identified in TR-34 and the RAI responses. Furthermore, the staff finds that these changes are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.4.1.5 Other Changes

In Revision 17 to the DCD, the applicant also proposed to change the following RCP design parameters in Table 5.4-1: (1) maximum continuous CCW inlet temperature; (2) motor/pump rotor minimum required moment of inertia; and (3) RCP estimated unit overall height and weight.

The applicant revised DCD Table 5.4-1 by adding Note 1, associated with the maximum continuous CCW inlet temperature of 95 °F, stating that an elevated CCW supply temperature of up to 110 °F may occur for up to 6 hours. In its response to RAI-SRP5.4.1-SRSB-02, the applicant explained that it intended Note 1 to clarify that although the design temperature of the CCW system cooling water to the RCP is 95 °F, transient conditions during normal plant cooldown or extreme climate events may occur during which the CCW system temperature may exceed 95 °F (but no more than 110 °F) for a short time. The staff concludes that the addition of Note 1 is acceptable.

In DCD Table 5.4-1, the applicant changed the motor/pump rotor minimum required moment of inertia from 16,500 lb-ft^2 to the phrase "Sufficient to provide flow coastdown as given in Figure 15.3.2-1." The applicant had previously proposed a similar change to the acceptance criterion of the RCP rotating inertia in DCD Tier 1, Table 2.1.2-4, "Inspections, Tests, Analyses, and Acceptance Criteria," to a specific pump coastdown curve specified in DCD Tier 1,

Figure 2.1.2-2, "Flow Transient for Four Cold Legs in Operation, Four Pumps Coasting Down," which is consistent with DCD Tier 2, Figure 15.3.2-1. The applicant proposed this change because the specified pump rotating inertia does not guarantee that the flows produced during pump coastdown will satisfy the analyses in the absence of information on pump resistance. The staff had found this change to be acceptable, as described in Supplement 1 to NUREG-1793. Therefore, the proposed change to Table 5.4-1 regarding pump rotor minimum required moment of inertia is made to be consistent with the four-pump coastdown curve in DCD Figure 15.3.2-1, which is used in the safety analysis of the design-basis transient of a complete loss of reactor coolant flow described in DCD Section 15.3.2; therefore, it is acceptable.

In Table 5.4-1, the applicant changed the RCP estimated unit overall height from 6.693 m (21 ft 11.5 in) to 6.706 m (22 ft), and it changed the total estimated weight of the motor and casing from 83688 kg (184,500 pounds mass (lbm)) to 90718 kg (200,000 lbm). These changes reflect the changes in the pump casing discharge nozzle and the use of the bi-metallic flywheel design.

In DCD Section 5.4.1.2.1, the applicant changed the RCP vibration monitoring system instrumentation from three-axis monitoring to five vibration monitors. In TR-34, the applicant stated that the instrumentation of the RCP is modified from the preliminary design to provide a more robust monitoring and diagnostic capability of the RCP. These changes include the addition of key phasors to aid in diagnostics in the event of high-vibration indications, and the addition of more vibration monitors to supply measurements in two planes at two different axial locations for diagnostic purposes. The staff concludes that these RCP instrumentation changes provide a more robust monitoring and diagnostic capability of the RCP without affecting RCP performance; therefore, they are acceptable.

In Revision 17 of DCD, the applicant updated Section 5.4.1.3.6.1 to specify the minimum damped natural frequency of the RCP rotating assembly as greater than 120 percent of the normal operating speed. The staff reviewed the various factors that were considered in determining the damped natural frequency of the RCPs. There is considerable energy dissipation from the effects of bearing films, can or winding annular fluid interaction, motor magnetic phenomena, and pump structure. The determination of the damped natural frequency of the RCP rotor bearing system model considered these effects on the damped natural frequency. The high degree of damping as a result of energy dissipation ensures a stable and smooth operation of the pump over a sufficiently wide range of pump operating speeds. The pumps have been analyzed for the response of the rotor and stator to external forcing functions. The analysis considered the support and connection of the pump to the SG and piping. The applicant evaluated the responses using criteria that included critical loads, stress deformation, and wear and displacement limits to establish the actual system critical speeds. The staff finds the evaluation criteria for determining the RCP damped natural frequency and critical speeds to be reasonable and acceptable. The staff also finds that there is high energy dissipation because of the various damping factors, which would ensure stable and smooth pump operation.

5.4.1.6 Conclusions

The staff has reviewed the proposed changes to the RCP design. Based on the evaluation described, the staff concludes that the AP1000 RCP design meets the requirement in GDC 10, "Reactor Design"; therefore, it is acceptable. The staff finds that the material specifications used and the preservice inspections provide reasonable assurance of the flywheel integrity, and

that the Curtiss-Wright report demonstrates that, if the flywheel assembly fails, the flywheel components will not penetrate the pump pressure boundary structures. Therefore, the revisions proposed by the applicant to AP1000 DCD, Section 5.4.1, meet the requirements of GDC 1 and 4 and ASME Code, Section III and are acceptable.

5.4.2 Steam Generators

5.4.2.1 Steam Generator Design

5.4.2.1.1 Summary of Technical Information

In Revision 17 to the AP1000 DCD, the applicant proposed changes to its SG design. In a letter dated November 29, 2006, the applicant submitted for staff review TR-35, Revision 0, "AP1000 Steam Generator Description Changes" (APP-GW-GLN-010), which provides the technical justification for the proposed SG design changes. Additional information regarding TR-35 appears in letters dated June 7, 2007, and September 7, 2007. The proposed design changes are described below.

Antivibration Bar

The original design of the antivibration bar is described in DCD Section 5.4.2.4.2 and detailed in DCD Figure 5.4-2. In Revision 17, the applicant revised the description by removing the words "wide strips of" from Section 5.4.2.4.2 and reconfiguring Figure 5.4-2 to show a general antivibration bar design.

Tube Expansion

DCD Sections 5.4.2.2, 5.4.2.4.1, and 5.4.2.4.2 describe the tube expansion process as a hydraulic expansion through the full depth of the tubesheet. The intent remains to achieve full depth expansion; however, the requirement to hydraulically expand through the full depth limits the manufacturing processes available for this expansion. In Revision 16, the applicant removed the reference to hydraulic expansion from DCD Sections 5.4.2.2, 5.4.2.4.1, and 5.4.2.4.2.

Primary Separator Design

The original design of the moisture separators is described as a 19.1cm (7.5-in) separator arrangement. The applicant determined that this design could not achieve the industry standard upper bound limit for moisture carryover (0.1 percent) and, therefore, changed the design to a 50.8-cm (20-in) separator arrangement. In Revision 17, the applicant revised Figure 5.4-2 to illustrate its typical 50.8-cm (20-in) separator arrangement.

Primary Separator Material

DCD Section 5.4.2.4.1 states, "Nickel-chromium-iron alloy in various forms is used for parts where high velocities could otherwise lead to erosion/corrosion. These include the nozzles on the feedwater ring, startup feedwater sparger, and some primary separator parts." In Revision 16 to the DCD, the applicant removed the phrase "and some primary separator parts."

Startup Feedwater Elevation

DCD Section 5.4.2.2 describes the startup feedwater nozzle elevation as being "just below" the main feedwater nozzle. In Revision 17 to the DCD, the applicant revised this to state that the startup feedwater nozzle elevation is the same as the main feedwater nozzle elevation.

Table 5.2-1 Correction

DCD Section 5.4.2.4.1 includes the table for the list of materials and is called "Table 5.2.3-1." In Revision 16 to the DCD, the applicant revised the table number to "Table 5.2-1."

Tube Location Identification

DCD Section 5.4.2.5 describes the number of tubes being scribed as "large." However, the space between the tubes and welds is not sufficient to allow for scribing a large fraction of the tubes. In Revision 17 to the DCD, the applicant removed the word "large" from Section 5.4.2.5.

5.4.2.1.2 Evaluation

GDC 32 requires, in part, that the designs of all components that are part of the reactor coolant pressure boundary permit periodic inspection and testing of critical areas and features to assess their structural and leak-tight integrity. The staff reviewed changes to this section as related to the proposed SG design changes.

In TR-35, the applicant proposed design changes to its SGs. Specifically, the applicant made minor description changes to the following: antivibration bars, tube expansion process, primary separator design, primary separator material, startup feedwater elevation, and tube location identification. The staff finds that these proposed changes meet the requirements of GDC 32 and are acceptable because they are consistent with current industry designs and practices and do not present a challenge to the integrity of the SG. Some of the changes were necessary to correct erroneous statements and inaccurate descriptions.

The NRC staff reviewed the proposed changes to the AP1000 DCD. The proposed changes establish the proposed design as the single, standard design for all AP1000 plants. These DCD changes are generic and are expected for all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plant. Thus, the proposed changes contribute to the increased standardization of the certification information in the AP1000 DCD.

5.4.2.1.3 Conclusions

Based on the above evaluation, the staff concludes that the SG design changes meet GDC 32 and are acceptable because they are consistent with current industry designs and practices and do not present a challenge to the integrity of the SG. Furthermore, the staff finds that the conclusions regarding the proposed SG design changes are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.4.2.2 Steam Generator Inservice Inspection

5.4.2.2.1 Summary of Technical Information

In Revision 17 to the AP1000 DCD, proposed changes to the AP1000 generic Technical Specification (TS) related to adopting Technical Specification Task Force Traveler (TSTF)-449, "Steam Generator Tube Integrity," Revision 4 of May 6, 2005. The current Westinghouse Owners Group Standard Technical Specifications (STS), NUREG-1431, "Standard Technical Specifications Westinghouse Plants," Revision 3.1, dated December 1, 2005, incorporates TSTF-449. These changes also relate to COL Information Item 5.4-1 identified in DCD Section 5.4.15. Implementation of TSTF-449 includes the following TS changes:

- clarification to the definition of identified leakage
- changes to TS 3.4.7, "Reactor Coolant System Operational Leakage," which modifies two condition statements and the two associated surveillance requirements
- addition of new TS 3.4.18, "Steam Generator (SG) Tube Integrity"
- modification of TS 5.5.4, which is renamed, "Steam Generator (SG) Program"
- modification of TS 5.6.8, "Steam Generator Tube Inspection Report"
- associated changes to the TS bases

5.4.2.2.2 Evaluation

GDC 32 requires, in part, that components that are part of the reactor coolant pressure boundary shall be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. The staff evaluated the applicant's proposed changes to the AP1000 generic TS as they relate to adopting TSTF-449 to ensure consistency with the latest revision of the STS using the acceptance criteria in NUREG-0800 Section 5.4.2.2, "Steam Generator (SG) Program." In addition, 10 CFR 50.55a(b)(2)(iii) states that if the TS include SG surveillance requirements that are different than those in Article IWB-2000 of ASME Code, Section XI, then the SG tube inspection requirements are governed by the TS.

Accordingly, the staff compared the proposed changes to the corresponding language in the STS (NUREG-1431, Revision 3.1). The staff determined that the changes are consistent with the approved STS and are appropriate to leakage and SG tube integrity requirements as they apply to the AP1000 standard plant design. Therefore, they are acceptable. With respect to tube integrity considerations, the Model Delta-125 SG planned for the AP1000 closely resembles the Model Delta-75 SGs installed as replacements at some operating plants. The staff also confirmed that Revision 17 to the DCD retains Section 5.4.15, which requires COL applicants to address SG tube integrity with a SG tube surveillance program and periodic monitoring of degradation of SG internals. .

5.4.2.2.3 Conclusions

Based on the evaluation above, the staff concludes that the changes proposed to the AP1000 TS for SG tube integrity are acceptable and meet the requirements of GDC 32. This conclusion is based on the consistency of the SG tube integrity program with the STS for the applicant's domestic PWRs. Furthermore, the staff finds that the conclusions regarding the changes to the TS for SG tube integrity will apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

5.4.4 Main Steam Line Flow Restriction

5.4.4.1 Summary of Technical Information

In Revision 17 of DCD Section 5.4.4, two changes are made regarding the main steam line flow restrictor.

- In Section 5.4.4.2, the material for the steam generator flow restrictor venturi inserts is changed from nickel-chromium-iron Alloy 600 (ASME 163) to nickel-chromium-iron Alloy 690.

- In Section 5.4.4.3, the pressure drop of the flow restrictor at 100 percent steam flow is changed from approximately 8.0 pounds per square inch (psi) to 15 psi.

5.4.4.2 Evaluation

For Section 5.4.4.2, since Alloy 690 is used in the AP1000 SG design, the change of the flow restrictor venturi insert material from Alloy 600 to Alloy 690 is consistent with the SG design and is acceptable.

For Section 5.4.4.3, the pressure drop change is based on the design analysis and is acceptable.

5.4.4.3 Conclusion

The staff has reviewed the proposed changes to DCD Section 5.4.4. Based on the above evaluation, the staff concludes that the proposed changes are acceptable.

5.4.5 Pressurizer

Section 5.4.5 of DCD Revision 17 refers to TR-36, "AP1000 Pressurizer Design" APP-GW-GLR-016 of May 2006. In TR-36, the applicant summarizes the design changes for the AP1000 pressurizer to accommodate the available space as a result of piping analysis. The pressurizer design changes preserve the same total internal volume of 59.5 cubic meter (m^3) (2100 cubic feet (ft^3)), while reducing the vessel height from 1541.8 cm (607 in) to 1277.6 cm (503 in) and increasing the vessel diameter from 228.6 cm (90 in) to 254.0 cm (100 in). Though DCD Revision 17, Table 5.1-2 reflects the dimensional change, Section 5.4.5 does not, because the total pressurizer volume did not change. However, DCD Revision 17, Section 15.0 did change, because the changes to the pressurizer vessel cross-section area and vessel height affect the pressurizer water level and setpoints of several functions of the reactor

protection system and engineered safety feature actuation systems. Chapter 15 of this SER addresses the effects on the transient and accident analyses.

5.4.7 Normal Residual Heat Removal System

5.4.7.1 Summary of Technical Information

The AP1000 RNS is a nonsafety-related system and is not required to operate to mitigate design-basis events. The primary functions of the RNS include the following: (1) remove decay heat and sensible heat from the core and the RCS; (2) provide LTOP; (3) provide a flow path for long-term, post accident makeup to the containment inventory; and (4) provide RCS and refueling cavity flow to the chemical and volume control system (CVS) for purification.

In Revision 17 to the DCD, the applicant proposed the following changes related to the RNS design:

- Revise the method of flow control through the RNS heat exchangers (HX) to improve the cooldown rate control.
- Provide auto close to V029, RNS motor-operated flow control valve to CVS, on high temperature to prevent potential damage to CVS demineralizer resin.
- Resize the RNS suction line relief valve to reduce valve instability.
- Increase ambient wet bulb temperature to 86.1 °F with zero percent exceedance to facilitate consideration of other potential nuclear plant sites.
- Relocate the RCS hot leg RTDs to provide required post accident monitoring data to upstream of the HX to indicate the RCS hot-leg temperature when in reduced inventory conditions.
- Revise the RNS long term makeup flow path to provide a simplified flow path for containment makeup through the manual containment isolation test connection valve in the discharge of the RNS.
- Revise Figure 5.4-7 to reflect containment penetration valve and piping changes.

5.4.7.2 Evaluation

The staff reviewed the changes to DCD Section 5.4.7 in accordance with NUREG-0800 Section 5.4.7, "Residual Heat Removal (RHR) System." The changes are acceptable if the system continues to satisfy the relevant requirements in NUREG-0800 Section 5.4.7.

In Section 5.4.7.4.2, "Plant Cooldown," an editorial change provides a description of the method used to control the cooldown rate as stated: "The cooldown rate is controlled by throttling the flow through the heat exchanger based on reactor coolant temperature." Flow through the RNS heat exchangers will be controlled with the HX discharge valves, V006A(B). This is accomplished by switching the control signals on valves V008A(B) with those on valves V006A(B). The NRC staff agrees that this method of control will reduce the probability of an operator error during a plant cooldown condition because this is the standard method of

controlling cooldown rate. Also, this revision is pictorially reflected on Figure 5.4-7 of the control signals to the valves. The NRC staff finds this change acceptable because it has the potential to improve operator performance by reducing the potential for operator error.

An additional revision to Figure 5.4-7 included a modification to V029, RNS motor-operated control valve downstream of the RNS HX connecting to the CVS. This change will allow the valve to auto close on high temperature. The reason for this change is to prevent high temperature fluid from passing through the demineralizers with the potential of damaging the resin. The resin in the CVS demineralizers can be damaged if the temperature through the demineralizers exceeds 60 °C (140 °F) for extended periods of time. The NRC staff finds this change acceptable because the change improves the reliability of the CVS system by reducing the potential damage to the resin.

In another revision to Figure 5.4-7, the RNS suction line relief valve is resized from a 10.2 cm (4 in) inlet and a 15.2 cm (6 in) outlet to a 7.6 cm (3 in) inlet and a 10.2 cm (4 in) outlet. RNS suction line relief valve resizing is implemented to reduce valve instability caused by the valve being fluid starved, which has the potential to cause valve chattering. The RNS relief valve is sized to mitigate primary overpressure events at low temperature conditions due to a heat injection and/or mass injection transient where the mismatch flow rate is no higher than 177 gpm. The staff finds this change acceptable because it improves the reliability of the valve by reducing potential damage to the valve due to unnecessary valve chatter.

The applicant modified DCD Section 5.4.7.1.2.1 to state that the CCW system supply temperature to the RNS HX is based on the maximum normal ambient wet bulb temperature as defined in Chapter 2, Table 2-1. The maximum normal ambient temperature is assumed for shutdown cooling. DCD Section 5.4.7.1.2.3 revises the maximum ambient design wet bulb temperature from 27.2 °C (81 °F) to 30.1 °C (86.1 °F) to facilitate consideration of other potential nuclear plant sites. The applicant has stated that the evaluation indicates there is sufficient margin in the systems design for the RNS, CCW system, and service water system (SWS) to maintain the same criteria and design basis with the increased ambient wet bulb temperature. In RAI-SRP5.4.7-SRSB-01, the staff requested that the applicant provide the results of the evaluation that demonstrate the RNS maintains sufficient margin. In its July 17, 2008, response, the applicant stated that "cooling the [in-containment refueling water storage tank] IRWST with the RNS, CCS, and SWS is not required to ensure that the plant can be maintained in a long-term safe condition." Therefore, under these circumstances, the staff agrees that the RNS, CCW system, and SWS designs would maintain sufficient margin with the revised higher ambient wet bulb temperature. The staff concludes that the proposed changes are acceptable because the analysis demonstrates that the RNS system can perform its intended function for normal operating conditions and is not required during transients since the RNS is not considered a safety-related system, i.e., no operational credit is assigned in Chapter 15, "Accident Analysis."

Section 5.4.7.2.1 was revised to clarify the temperature instrumentation employed to monitor the RCS hot leg for RNS operation during different RCS level conditions. For normal RCS inventory conditions, the RCS hot-leg wide-range temperature instruments are monitored; whereas, during reduced RCS inventory conditions, the RNS temperature instruments located upstream of the HX are monitored for the RCS hot-leg temperature. The staff finds that the revision is an editorial clarification with no impact on the RNS functionality. Therefore, the staff concludes the revision is acceptable.

Safety-related makeup water can be provided through the RNS for long-term post accident containment makeup. DCD Revision 17 changed Section 5.4.7.5 to state that this makeup is provided through the manual containment isolation test connection valve in the discharge of the RNS. This change simplifies the long-term makeup flow path setup by eliminating a component in the flow path, improves the RNS long-term makeup flow method, and has no impact on the functionality of the RNS. Therefore, the staff concludes that the change is acceptable.

In Tier 1, Table 2.3.6-1, the applicant deleted RNS HX A and B channel head drain valves RNS-PL-V046A and B and added RNS discharge containment isolation test connection RNS-PL-V012 to reflect the new RNS long-term makeup flow path to provide containment makeup. It also revised Figure 2.3.6-1 to reflect this change to the long-term makeup flow path configuration.

5.4.7.3 Conclusions

The staff reviewed the proposed changes to DCD Section 5.4.7 and concludes that the proposed changes improve overall system performance with the potential for improved operator performance without having a significant effect on the RNS design basis or RNS interfacing systems and meet the relevant acceptance criteria of NUREG-0800 Section 5.4.7.

6. ENGINEERED SAFETY FEATURES

Westinghouse has submitted information in support of its design certification (DC) amendment application that Westinghouse (the applicant) considers "proprietary" within the meaning of the definition provided in Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390(b)(5), "Public inspections, exemptions, requests for withholding." The applicant has requested that this information be withheld from public disclosure and the Nuclear Regulatory Commission (NRC) staff agrees that the submitted information sought to be withheld includes proprietary commercial information and should be withheld from public disclosure. This chapter of the NRC staff's evaluation includes proprietary information that has been redacted in order to make the evaluation available to the public. The redacted information appears within "square brackets" as follows:

[ ]

The complete text of this chapter, including proprietary information can be found at Agencywide Documents Access and Management System (ADAMS) Accession Number ML112091879 and can be accessed by those who have specific authorization to access Westinghouse proprietary information.

6.1.1 Metallic Materials

6.1.1.1 Summary of Technical Information

The applicant revised Section 6.1.1.2 of the AP1000 design control document (DCD) Tier 2 to reflect a proposed modification to DCD Tier 2, Table 5.2-2, "Reactor Coolant Water Chemistry Specifications," to allow the injection of zinc into reactor coolant water. The basis for this change appeared in Westinghouse Electric Company, LLC, technical report (TR)-32, APP-GW-GLN-002, "AP1000 Licensing Design Change Document Zinc Addition," dated April 5, 2006. The proposed change to Section 6.1.1.2 deletes the reference to zinc as a material that is not allowed to come into contact with engineered safety feature (ESF) components made of stainless steel. In Revision 17 of the DCD, the applicant changed Section 6.1.1.2 to make it consistent with Table 5.2-2.

The applicant revised DCD Tier 2, Section 6.1.1.3, to replace American Society for Testing and Materials (ASTM) A240 Type XM-29 with ASTM/American Society of Mechanical Engineers (ASME) A240/SA-240 UNS S32101 (LDX 2101) for the fabrication of the in-containment refueling water storage tank (IRWST). Westinghouse's TR-106, APP-GW-GLN-106, "AP1000 Standard Combined License Technical Report, AP1000, Licensing Design Changes for Mechanical System and Component Design Updates," Revision 1, dated September 28, 2007, identified and justified this change. Westinghouse's TR-134, "AP1000 DCD Impacts to Support COLA Standardization," Revision 5, dated June 27, 2008, details the modifications pertaining to DCD Section 6.1.1.3. DCD Tier 2, Revision 17, incorporates the proposed modifications.

In addition, the applicant revised DCD Tier 2, Section 6.1.1.4, to add a design requirement that aluminum excore detectors be enclosed in stainless steel or titanium to address, in part, Generic Safety Issue (GSI)-191, "Assessment of Debris Accumulation on PWR Sump Performance." The applicant also modified DCD Tier 1, Table 2.2.3-4, "Inspections, Tests, Analyses, and Acceptance Criteria," to require verification that the excore detector surfaces are

made of stainless steel or titanium. TR-134 details the modifications to the DCD. The justification for these modifications appears in Westinghouse's TR-26, APP-GW-GLR-079, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA," Revision 7, dated February 26, 2010, and APP-GW-GLE-002, "Impacts to the AP1000 DCD to Address Generic Safety Issue (GSI)-191," Revision 7, dated July 13. DCD Tier 1 and 2, Revision 17, incorporates the proposed modifications.

6.1.1.2 Evaluation

DCD Tier 2, Section 6.1.1.2, states that lead, antimony, cadmium, indium, mercury, zinc, and tin metals and their alloys are not allowed to come into contact with ESF component parts made of stainless steel or high-alloy metals during fabrication or operation. The applicant proposed to modify DCD Tier 2, Section 6.1.1.2, to delete zinc as a material that is not allowed to come into contact with ESF components made of stainless steel. This proposed modification results from a proposed change in the reactor coolant water chemistry specifications, detailed in Table 5.2-2, to allow the injection of zinc into reactor coolant water. The basis for this proposed change relates to the benefits resulting from the addition of zinc to primary coolant which reduces radiation fields and the formation of crud. Section 5.2.3 of this safety evaluation report (SER) includes a detailed staff evaluation of the applicant's proposed change related to the addition of zinc to reactor coolant, which the staff finds acceptable. This proposed modification is generic and is expected to apply to all combined license (COL) applications referencing the AP1000 DC. Therefore, the proposed DCD change is acceptable pursuant to 10 CFR 52.63(a)(1)(vii), "Finality of standard design certifications," on the basis that it contributes to the increased standardization of the certification information.

DCD Tier 2, Section 6.1.1.3, describes the materials for nonpressure-retaining portions of ESFs in contact with borated water or other fluids. The IRWST liner and the passive containment cooling system (PCS) storage tank liner are primary examples of these items.

General Design Criterion (GDC) 4, "Environmental and Dynamic Effects Design Basis," of Appendix A, "Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Domestic licensing of production and utilization facilities," requires that structures, systems, and components (SSCs) important to safety be designed to accommodate the effects of, and to be compatible with, the environmental conditions associated with normal operations, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). In order for the IRWST to meet the requirements of GDC 4, the materials used must be compatible with ESF fluids.

Currently, AP1000 DCD, Section 6.1.1.3, states that ASTM A240 Type XM 29 (Nitronic 33) may be used to fabricate the IRWST. Table 6.1-1 specifies that XM-29 or TP304 will be used to fabricate the IRWST. The applicant proposed to modify the stainless steel surfaces of the containment internal structural modules as part of design change 049, which is described in TR-106. This design change will affect the IRWST, which is part of the ESFs. The proposed change to Section 6.1.1.3 specifies ASTM/ASME A240/SA240 UNS S32101, commonly known as LDX 2101, for fabrication of the IRWST.

The applicant provided its basis for the selection of LDX 2101 in TR-106, which stated that the proposed change to use LDX 2101 is because of the limited availability of Nitronic 33 in the required plate sizes. The applicant also stated that LDX 2101 has similar corrosion resistance and is easily welded.

LDX 2101 is an austenitic-ferritic (duplex) stainless steel. LDX 2101 provides adequate resistance to uniform corrosion, pitting, crevice corrosion, and stress-corrosion cracking in most media. It has higher mechanical strength than traditional stainless steels such as 304, 316L, and XM-29. LDX 2101 can be readily welded with commonly used welding processes. LDX 2101 is a relatively new material that was adopted in ASTM SA-240 in 2007. Its current commercial uses range widely from chemical storage tanks, waste water treatment facilities, and over-the-road chemical transportation tanks. This material is now widely used in several industries as a replacement for grade 304 and 316 stainless steels. The information provided by the applicant and the manufacturer's literature suggest that this material will perform adequately when exposed to borated water environments. Duplex stainless steels are considered to have more than adequate resistance to general corrosion and stress-corrosion cracking in aqueous solutions containing boric acid and chlorides.

A material's resistance to pitting can be compared to other materials using its pitting resistance equivalent (PRE) number. The PRE number is a theoretical method to compare the resistance to pitting and crevice corrosion of different types of stainless steels based on their chemical composition. In response to request for additional information (RAI)-TR106-CIB1-04, dated January 29, 2008, the applicant provided PRE numbers for stainless steel materials 304, 316L, and XM-29. The applicant also provided PRE numbers for LDX 2101, 2304 duplex stainless steel, and 2205 duplex stainless steel. The information provided by the applicant suggests that LDX 2101 has improved resistance to pitting and crevice corrosion when compared to 304, 316L, and XM-29 stainless steels. The applicant provided data in its January 29, 2008, letter that indicates that duplex stainless steels perform better than or equal to austenitic stainless steels, such as Types 304 and 316L, when exposed to borated water and chlorides. In addition, the applicant stated that it was just beginning its corrosion testing program for LDX 2101 base material and LDX 2101 welds. The staff notes that the information supplied by the applicant included data for duplex stainless steels 2304 and 2205 but did not include data for LDX 2101. While the staff agreed that LDX 2101 was likely to perform in a manner similar to 2304 and 2205 duplex stainless steels in borated water, confirmatory testing must be completed before the staff can approve the use of this material.

In supplemental RAI-TR106-CIB1-05, the staff asked the applicant to discuss its corrosion test plan and acceptance criteria for LDX 2101 base material and LDX 2101 weld filler materials. In addition, the staff requested that the applicant provide a technical justification for its testing plan and acceptance criteria that describe its adequacy to ensure that the materials will not be subject to general corrosion, stress-corrosion cracking, or other form of degradation from corrosion for the life of the plant.

The applicant responded to RAI-TR106-CIB-05 in a letter dated May 14, 2008. The applicant stated that it would conduct a confirmatory corrosion testing program to demonstrate the adequacy of LDX 2101. The applicant's confirmatory corrosion testing program includes LDX 2101 base material and weld filler materials that bound those filler materials that will be used during fabrication. Tests that will be conducted include uniform corrosion, stress-corrosion cracking, and crevice corrosion tests. The applicant stated that its test program was designed to establish test data on LDX 2101 material and its welds in terms of their susceptibility to degradation under exposure to oxygenated boric acid with halogen (chloride) contamination and in crevice corrosion conditions under accelerated service conditions to demonstrate a service life of 60 years. All tests will use Type 304 austenitic stainless steel as a reference sample. The staff notes that it approved Type 304 stainless steel, in addition to XM-29, for use in the fabrication of the IRWST liner in NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," issued September 2004.

The staff found that the information that the applicant provided suggests that LDX 2101 duplex stainless steel material is likely to perform adequately in the IRWST environment. However, given the lack of specific corrosion data for LDX 2101, the staff asked, in RAI-SRP6.1.1-CIB1-02, that the applicant provide the results from its LDX 2101 corrosion testing program and describe the extent to which the results confirm that LDX 2101 will not be subject to general corrosion, stress corrosion, or other forms of degradation for the design life of the plant. The applicant responded in a letter dated June 30, 2010, and stated that it has completed a confirmatory testing program that included LDX 2101 and its associated welds under service conditions in the AP1000 IRWST. The test program included testing for three different types of corrosion: uniform corrosion, crevice corrosion, and stress-corrosion cracking. The testing also included Type 304 L stainless steel samples as reference samples for benchmarking. The staff notes that it approved Type 304 austenitic stainless steel for use in the IRWST, as documented in NUREG-1793. The applicant stated that the overall results of its confirmatory tests demonstrate that S32101 (LDX 2101) duplex stainless steel and its welds exhibited superior performance under accelerated service conditions in comparison with 304L stainless steel for use in the AP1000 structural modules [ ]. The staff notes that Types 304 and 304L are commonly used materials for applications such as the IRWST in operating plants. Westinghouse Commercial Atomic Power (WCAP)-17280-P, "Confirmatory Corrosion Testing of S32101/LDX2101 Duplex Stainless Steel Base and Weld Materials for the AP1000 Structural Module Applications," issued June 2010, documents the test program and the results. The staff conducted an audit of WCAP-17280-P at Westinghouse's Rockville, Maryland, office on June 28, 2010. As a result of the audit, the staff concluded that the test data in WCAP-17280-P provide reasonable assurance that LDX 2101 will not be subject to general corrosion, stress corrosion, or other forms of degradation for the design life of the plant and is, therefore, acceptable.

The staff finds that the proposed modifications to Section 6.1.1.3 are acceptable and meet the requirements of GDC 4 and the acceptance criteria of NUREG-0800, "Standard Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants," Section 6.1.1, "Engineered Safety Features Materials," Revision 2, issued March 2007. Revision 17 of the DCD and TR-134, Revision 5, incorporate the proposed changes as identified in TR-106. Accordingly, these changes are generic and are expected to apply to all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plants. Thus, the proposed changes contribute to the increased standardization of the AP1000 certified design and, therefore, meet the requirements of 10 CFR 52.63(a)(1)(vii).

DCD Tier 2, Section 6.1.1.4, describes the materials' compatibility with reactor coolant and ESF fluids. Section 6.1.1.4 currently states that in the postaccident environment, both aluminum and zinc surfaces in the containment are subject to chemical attack, resulting in the production of hydrogen. The applicant proposed to modify Section 6.1.1.4 to state that in the postaccident environment, both aluminum and zinc surfaces in the containment are subject to chemical attack, resulting in the production of hydrogen or chemical precipitants or both that can affect long-term core cooling. Primary sources of aluminum in the AP1000 containment are the excore detectors. To avoid sump water contact with the excore detectors, the applicant proposed to modify the DCD to state they will be enclosed in stainless steel or titanium housings. In addition, the applicant has proposed to modify DCD Tier 1, Table 2.2.3-4, to include inspections, tests, analyses, and acceptance criteria (ITAAC) to verify that exposed surfaces of the excore detectors are made of stainless steel or titanium. The applicant's basis for the proposed change appears in Westinghouse DCD Impact Document and TR-26. These proposed modifications are intended to address, in part, GSI-191. Section 6.2.1.8 of this report

provides a detailed staff evaluation of the applicant's design related to GSI-191. The staff finds the applicant's proposal to enclose the excore detectors in stainless steel or titanium acceptable because these materials will prevent the degradation of the aluminum excore detectors, which could result in chemical precipitants that can affect long-term core cooling. Revision 17 of the DCD incorporates the proposed changes, as identified above. Accordingly, these changes are generic and are expected to apply to all COL applications referencing the AP1000 certified design. At this time, the NRC has not issued a COL for any AP1000 plants. Thus, the proposed changes contribute to the increased standardization of the AP1000 certified design and, therefore, meet the requirements of 10 CFR 52.63(a)(1)(vii).

6.1.1.3 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the AP1000 DCD, Section 6.1.1, are acceptable. Revision 17 to the AP1000 DCD incorporates the proposed changes, as identified in TR-134, TR-32, TR-26, TR-106 and the AP1000 DCD Impact Document. Furthermore, the staff finds that the conclusions in the above supporting documentation on the evaluation of these proposed DCD modifications are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

6.1.2 Organic Materials

6.1.2.1 Summary of Technical Information

The DCD changes described below are based on DCD Revision 17 and Westinghouse document APP-GW-GLE-002, Revision 7 (DCD Impact Document). The applicant modified DCD Tier 2, Section 6.1.2.1, in several places to change or clarify the types of coatings, the locations where they are used, and the associated quality assurance requirements.

The staff evaluated coatings for the certified AP1000 design based on NUREG-0800 Section 6.1.2. The staff concluded that the design met the quality assurance requirements of Appendix B to 10 CFR Part 50, as they relate to protective coatings. This conclusion was based on the design's conforming to the guidance in regulatory guide (RG) 1.54, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," Revision 1, including the provisions for design basis accident testing. RG 1.54, Revision 1, also provides guidance for coatings application and assessment.

The changes in DCD Revision 17 and the DCD Impact Document were proposed in order to meet the relevant requirements of GDC 35, "Emergency Core Cooling," GDC 38, "Containment Heat Removal," and 10 CFR 50.46(b)(5), "Acceptance criteria for emergency core cooling systems for light-water nuclear power reactors," for cooling water following a LOCA. Coatings are discussed in that context in Section 6.3 of the AP1000 DCD and 6.2 of the staff's corresponding safety evaluation. The proposed changes include the COL information required in Section 6.1.3.2 for consistency with Section 6.1.2. Additional information was provided in TR-26.

6.1.2.2 Evaluation

Inside containment, as described in the revised Section 6.1.2.1.5, inorganic zinc (IOZ) will be applied only to surfaces that exceed the temperature limit for epoxy during normal operation. All

IOZ inside containment will be classified as Service Level I. The staff finds these changes acceptable, since appropriately qualified epoxy coatings are suitable replacements for corrosion protection at lower temperatures, and because the Service Level I designation conforms to RG 1.54, Revision 1. Section 6.1.2.1.2 also introduces self-priming high-solids epoxy (SPHSE) without a zinc primer for structural modules, and the zinc/epoxy combination is eliminated except for the inside of the containment vessel up to seven feet above the operating floor. SPHSE is also added as a possible coating on concrete floors and walls. The staff finds this acceptable because the technical and quality requirements relating to the epoxy selection, and conformance to RG 1.54, Revision 1, are not changed. SPHSE is a standard type of coating, and SPHSE products have been design basis accident (DBA) qualified for steel and concrete. Eliminating the zinc/epoxy combination (except on the containment shell near the operating floor) is acceptable because there is no change in the quality requirements for the SPHSE.

Another significant change for inside containment is that coatings on manufactured components below the LOCA flood zone or in locations susceptible to debris transport must have a density of at least 1602 kilograms per cubic meter (kg/m^3) (100 pounds per cubic foot (lb/ft^3)) or a report, approved by the NRC, demonstrating the debris from that coating will not transport. This is discussed in DCD Sections 6.1.2.1.5 and 6.1.2.1.6. The staff finds this acceptable because the high density requirement limits transport of potential debris, while other coating requirements are not changed. SPHSE was also introduced as a coating for outside containment. According to the modifications in Section 6.1.2.1.4, carbon steel outside containment will be coated with either SPHSE or IOZ with epoxy top coat. Concrete floors and walls will be coated with epoxy or SPHSE. Section 6.1.2.1.6 was changed to clarify that procurement of Service Level II coatings outside containment, unlike inside containment, is not a safety-related, 10 CFR Part 50, Appendix B activity.

DCD Section 6.1.3.2, the COL information item called "Coating Program," was modified to state that the COL applicant must include Service Level II coatings in the coatings program, and the coatings program includes inspection along with procurement, application, and monitoring. (This is also identified in the DCD as COL Information Item 6.1-2.) As indicated in Tier 2 Appendix 1A of the DCD, there is an exception to RG 1.54, Revision 1 due to the use of coatings inside containment that are designated Service Level II. The quality assurance requirements for these coatings are discussed in DCD Section 6.1.2.1.6. The change to Section 6.1.3.2 was proposed in a March 31, 2010, response to RAI-SRP6.1.2-CIB1-01. The staff found it acceptable because it makes the COL information required by Section 6.1.3.2 consistent with the information in Section 6.1.2.1.6. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Tier 1, Table 2.2.3-4, Item "x," was modified to add two requirements to the existing requirement for nonsafety-related coatings used inside containment on walls, floors, ceilings, and structural steel (except for inside a chemical and volume control system (CVS) room that drains to the waste liquid processing system). The existing requirement (acceptance criterion) is a report concluding that these coatings have a dry film density of at least 1602 kg/m^3 (100 lb/ft^3). The two additional acceptance criteria are for a report showing that these coatings will not transport if the density is less than 1602 kg/m^3 (100 lb/ft^3) and for a report concluding that inorganic zinc coating used on these surfaces is safety Service Level I. (This ITAAC includes components in the LOCA flood zone, or above the flood zone and not in an enclosure.) The staff finds these changes acceptable because they supplement the existing ITAAC intended to ensure coatings chips generated by a LOCA will not transport.

6.1.2.3 Conclusion

Based on the above evaluation, the staff finds that the revisions proposed by the applicant to the AP1000 DCD, Section 6.1.2 are acceptable. The proposed changes are incorporated in Revision 17 of the DCD and in the AP1000 DCD impact document, APP-GW-GLE-002, Revision 7. Furthermore, the staff finds that conclusions about these DCD modifications in the documents cited above are generic and are expected to apply to all COL applications referencing the AP1000 DC. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

6.2.1 Primary Containment Functional Design

6.2.1.1 Containment Pressure and Temperature Response to High-Energy Line Breaks

6.2.1.1.1 Wet Bulb Temperature

6.2.1.1.1.1 Summary of Technical Information

As described in APP-GW-GLE-036, "Impact of a Revision to the Current Wet Bulb Temperature Identified in Table 5.0-1 (Tier 1) and Table 2-1 (Sheet 1 of 3) of the DCD (Revision 16)," Revision 0, issued June 27, 2008, the site parameters for external wet bulb temperatures are increased from 26.7 °Celsius (C) to 30 °C (80 °Fahrenheit (F) to 86.1 °F) coincident and 29.7 °C to 30 °C (85.5 °F to 86.1 °F) noncoincident to encompass more sites in the eastern United States. The change to the coincident wet bulb temperature corresponds to an increase in relative humidity from 22 percent to 31 percent at the maximum dry bulb temperature of 46.1 °C (115 °F) and atmospheric conditions. The change to the noncoincident wet bulb temperature increases the temperature at which 100 percent relative humidity can occur. The external temperature and relative humidity are initial conditions in the containment analysis.

6.2.1.1.1.2 Evaluation

In response to RAI-SRP6.2.1.1-SPCV-05, dated October 1, 2008, the applicant stated that the containment analysis results are not sensitive to the external relative humidity. The staff audited one of the supporting analyses, Appendix A.3 to APP-GW-GSC-040, "AP1000 WGOTHIC Containment Models for Integrated Safety Analysis Evaluation: Disposition of Design Change Proposals and Modification of the Containment Model," and observed that when the relative humidity of the NRC-approved AP1000 WGOTHIC model for a double-ended cold-leg guillotine (DECLG) LOCA was increased from 22 percent to 31 percent at an external temperature of 46.1 °C (115 °F), the resulting peak pressure increase was negligible. This study also demonstrated that the increase to noncoincident wet bulb temperature is less limiting than the increase to coincident wet bulb temperature with respect to peak pressures. Additional sensitivity studies documented in Section 5.6 of WCAP-15846, "WGOTHIC Application to AP600 and AP1000," Revision 1, issued March 2004, showed that the AP600 WGOTHIC LOCA and main steamline break models are insensitive to external humidity for select initial conditions. The staff ran confirmatory analyses using the CONTAIN model of the AP1000 containment developed by the staff during the DCD review. When the outside relative humidity in this model was increased to 31 percent, there was negligible impact on the peak pressures resulting from a main steamline break and DECLG. There was also a negligible increase in the pressure 24 hours after the DECLG LOCA.

While the staff found the increase to external wet bulb temperatures acceptable, it was not clear how the containment analyses referenced in the DCD would incorporate this change. In its July 12, 2009, Revision 2 response to RAI-SRP6.2.1.1-SPCV-06, the applicant stated that DCD changes to reference a more recent WGOTHIC model, which includes the increase to external wet bulb temperature, will be included in its submittal of APP-GW-GLR-096, "Evaluation of the Effect of AP1000 Enhanced Shield Building Design on the Containment Response and Safety Analysis." Revision 1 of APP-GW-GLR-096 was issued August 2010, and it does propose adding APP-GW-GLR-096 as a reference to the DCD. The staff confirmed that the model used in this study includes the increased wet bulb temperature, and is satisfied with the response. The evaluation of APP-GW-GLR-096 is documented in Chapter 23.

6.2.1.1.1.3 Conclusion

Based on the evaluations by the applicant and the confirmatory analysis by the staff, the staff concludes that that the containment functional design capability is essentially unchanged by the proposed increase in maximum site wet bulb temperatures. The conclusions reached in NUREG-1793, Sections 6.2.1 and 6.2.1.1, remain applicable, including that the design is compliant with regulatory requirements.

6.2.1.1.2 External Pressure Analysis

6.2.1.1.2.1 Summary of Technical Information

In DCD Revision 15, the maximum external pressure event is alleviated by the operator action of opening either set of purge valves. On August 16, 2010, the applicant submitted Change Number 74 to add a vacuum relief system that replaces the operator action. The "NRC Review Package" attached to this letter includes the revised analysis and proposed DCD changes for Section 6.2.1.1.4.

6.2.1.1.2.2 Evaluation

The staff evaluation of the adequacy of the vacuum relief system to mitigate the maximum expected external pressure scenario as described in Change Number 74 will be included in Chapter 23.

6.2.1.1.2.3 Conclusion

The staff's conclusion regarding the external pressure analysis is provided in Chapter 23.

6.2.1.2 Subcompartment Analysis

6.2.1.2.1 Summary of Technical Information

APP-GW-GLR-016, "AP1000 Pressurizer Design," issued May 2006, describes changes made to the diameter and height of the pressurizer and to the wall heights of the pressurizer compartment in order to obtain satisfactory piping analysis results. As a result of the shorter pressurizer walls, the applicant decreased the upper elevation of the assumed pressurizer spray break in DCD Tier 2, Section 6.2.1.2.3.2, from 52.1 to 49.7 meters (m) (171 to 163 feet (ft)).

6.2.1.2.2 Evaluation

The subcompartment analysis previously approved by the staff used the TMD computer code with models described in WCAP-15965, "AP1000 Subcompartment Models," issued November 2002. As reported in the July 18, 2008, response to RAI-SRP6.2.1.2-SPCV-01, the applicant performed a conservative calculation to evaluate the impact of the pressurizer changes. The results showed that the differential pressure (dP) in the pressurizer cubicle remained below the 34.5 kilopascal (kPa) (5 pounds per square inch differential (psid)) structural threshold. The staff ran a confirmatory analysis using COMPARE with input compiled from the pressurizer compartment model described in WCAP-15965 and the changes to the pressurizer described in APP-GW-GLR-016. The staff analysis predicted that the pressurizer changes would increase the maximum dP from [
], which is consistent with the applicant's results.

The revised height of the pressurizer wall is 48.8 m (160 ft). Based on the July 18, 2009, response to RAI-SRP6.2.1.2-SPCV-02, the pressurizer spray line extends 0.9 m (3 ft) above the top of the wall; therefore, changing the upper elevation for the assumed pressurizer spray break to 49.7 m (163 ft) is appropriate.

While the staff found the changes to the pressurizer and pressurizer compartment acceptable with respect to subcompartment analyses, it was not clear how the revised analysis would be incorporated into the DCD. In a letter dated August 31, 2009, in response to RAI-SRP6.2.1.2-SPCV-01 Revision 2, the applicant revised the DCD to state that the impact of the dimensional changes to the pressurizer and pressurizer compartment on the subcompartment analysis was evaluated in APP-GW-GLR-138, "Evaluation of the Pressurizer Changes on the AP1000 TMD Analyses," issued August 2009, and the existing conclusions remain valid. The staff is satisfied with this response. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

6.2.1.2.3 Conclusion

Based on the results of the applicant's analysis and the staff's confirmatory calculations, the staff agrees that the changes made in the DCD related to the pressurizer compartment have a negligible impact on the AP1000 subcompartment analysis. The conclusions reached in NUREG-1793, Section 6.2.1.2, remain applicable, including that the containment subcompartment pressurization analysis is acceptable.

6.2.1.3 Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents

6.2.1.3.1 Summary of Technical Information

While the applicant did not change DCD Section 6.2.1.3.2.1 regarding the 1 percent full-power allowance for calorimetric error in the energy release calculations, the applicant's response to RAI-SRP15.0-SRSB-02 proposes a change to this section as discussed below.

6.2.1.3.2 Evaluation

For analyses of heat sources during a postulated LOCA, paragraph I.A of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50 permits an assumed power level allowance of less than 2 percent full power if it has been demonstrated to account for uncertainties related to power level instrumentation error. The power level used to determine the maximum containment

pressure in Chapter 6 of the DCD was based on a calorimetric error of less than 2 percent full power, but there was no associated justification. In a May 6, 2009, response to RAI-SRP15.0-SRSB-02, the applicant stated that the AP1000 will use the proven technology of high-accuracy instrumentation to demonstrate a 1 percent design uncertainty and added COL Information Item 15.0-1, described in DCD Section 15.0.15, to track this commitment. The applicant also added a note to the DCD Section 6.2.1.3.2.1 assumptions on energy release to reference this COL item. This is appropriate as it provides justification for the uncertainty value, as required by the stated criteria.

6.2.1.3.3 Conclusion

The change is acceptable, and the conclusions reached in NUREG-1793, Sections 6.2.1.3 and 6.2.1.4, remain applicable, including that the methods and assumptions are acceptable for the licensing analyses.

6.2.1.8 Adequacy of In-Containment Refueling Water Storage Tank and Containment Recirculation Screen Performance

DCD Tier 2, Section 6.3.2.2.7, describes the evaluation of the water sources for long-term recirculation cooling following a LOCA, including the design and operation of the AP1000 passive core cooling system (PXS) debris screens. DCD Section 6.3.8 describes the associated COL information items, and DCD Tier 1, Section 2.2.3, includes the associated design descriptions and ITAAC. DCD Revision 17 incorporated many changes to these sections, as did APP-GW-GLE-002, which identified changes beyond those included in DCD Revision 17. Because the revisions are so extensive, the staff will not address each change independently but will perform a complete evaluation of the final configuration. As such, this section of the SER amendment replaces the analysis documented in NUREG-1793, Revision 0, in its entirety. In a subsequent revision to the AP1000 DCD, the applicant incorporated the DCD text proposed in APP-GW-GLE-002, Revision 7, dated July 13, 2010.

One of the changes made is the closure of COL Information Item 6.3-2 from DCD Tier 2, Table 1.8-2. DCD Tier 2, Section 6.3.8.2, "Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA," originally made the following commitment:

> The Combined License applicants referencing the AP1000 will perform an evaluation consistent with RG 1.82, Revision 3, and subsequently approved NRC guidance, to demonstrate that adequate long-term core cooling is available considering debris resulting from a LOCA together with debris that exists before a LOCA. As discussed in DCD Subsection 6.3.2.2.7.1, a LOCA in the AP1000 does not generate fibrous debris due to damage to insulation or other materials included in the AP1000 design. The evaluation will consider resident fibers and particles that could be present considering the plant design, location, and containment cleanliness program. The determination of the characteristics of such resident debris will be based on sample measurements from operating plants. The evaluation will also consider the potential for the generation of chemical debris (precipitants). The potential to generate such debris will be determined considering the materials used inside the AP1000 containment, the post-accident water chemistry of the AP1000, and the applicable research/testing.

In DCD Revision 17, the applicant stated the following:

The Combined License information item requested in this subsection has been fully addressed in APP-GW-GLR-079 (Reference 3), and the applicable changes are incorporated into the DCD. The design of the recirculation screens is complete. Testing to assess the screen performance and downstream effects is complete. A study of the effects of screen design and performance on long-term cooling is complete. No additional work is required by the Combined License applicant to address the aspects of the Combined License information requested in this subsection.

The following Commission regulations are related to the evaluation of the water sources for long-term recirculation cooling following a LOCA:

- GDC 35, as it relates to providing abundant emergency core cooling to transfer heat from the reactor core following a LOCA

- GDC 38, as it relates to the ability of the containment heat removal system to rapidly reduce the containment pressure and temperature following a LOCA and to maintain these indicators at acceptably low levels

- 10 CFR 50.46(b)(5), as it relates to requirements for long-term cooling in the presence of LOCA-generated and latent debris

As directed by NUREG-0800 Section 6.2.2, "Containment Heat Removal Systems," Revision 5, the staff performed the review in accordance with RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident," Revision 3, issued November 2003, as supplemented by the Nuclear Energy Institute (NEI) Guidance Report NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology," Revision 0, Volume 1, issued December 2004, and the associated NRC safety evaluation, "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to NRC Generic Letter 2004-02," issued December 2004. The review was also informed by WCAP-16406-P, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1, dated August 2007, as supplemented by "Final Safety Evaluation for Pressurized Water Reactor Owners Group Topical Report WCAP-16406-P," "Evaluation of Downstream Sump Debris Effects in Support of GSI-191, Revision 1," Revision 0, dated December 20, 2007; the NRC letter dated March 28, 2008, "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02," with enclosures addressing the areas of chemical effects, coatings, and head loss testing; the final safety evaluation by the Office of Nuclear Reactor Regulation on TR WCAP-16530-NP, "Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191," dated December 21, 2007; and the NRC letter dated April 6, 2010, "Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02."

In addition to the DCD, as modified by APP-GW-GLE-002, the staff reviewed APP-GW-GLR-079-P and APP-GW-GLR-086-NP (TR-26), "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA," Revision 8, dated July 20, 2010, and APP-GW-GLN-147-P and -NP (TR-147), "AP1000 Containment Recirculation and IRWST Screen Design," Revision 3, issued November 2009, which are identified as DCD references. The following submittals from the applicant provided additional information: WCAP-16914-P and -NP, "Evaluation of Debris Loading Head Loss Tests for AP1000 Recirculation Screens and In-Containment Refueling Water Storage Tank Screens,"

Revision 5, issued June 2010; WCAP-17028-P and -NP, "Evaluation of Debris Loading Head Loss Experiments Across AP1000 Fuel Assemblies During Post-Accident Recirculation," Revision 6, issued June 2010; APP-PXS-GLR-001, "Impact on AP1000 Post-LOCA Long-Term Cooling of Postulated Containment Sump Debris," Revision 4, dated February 26, 2010; APP-GW-GLR-092-P and APP-GW-GLR-093-NP, "Statistical Evaluation of AP1000 Fuel Assembly Debris-Loading Head loss Tests," Revision 0, issued February 2010; and APP-GW-GLR-110-P and APP-GW-GLR-111-NP, "Boric Acid Precipitation Tests During Post-LOCA Conditions," Revision 0, issued February 2010.

The applicant responded to staff RAIs in letters dated November 6 and 11, 2008; April 22, May 12, May 13, May 14, May 15, May 20, May 27, June 4, July 7, July 22, July 31, September 17, September 22, and November 2, 2009; and January 29, February 26, March 12, March 26, April 1, April 16, April 26, April 29, May 11, May 13, May 28, June 14, June 28, June 30 and July 30, 2010.

6.2.1.8.1 Summary of Technical Information

The AP1000 has two containment recirculation screens and three IRWST screens to capture debris following a LOCA. All screens are designed to be vertically oriented and have solid top covers so that debris that settles out of the water does not fall on the screening surfaces. The screens are constructed of stainless steel to be corrosion resistant and comprise individual pockets to provide greater filtering areas for a given volume. All AP1000 screens use the same pocket geometry, with each pocket having a frontal face area greater than or equal to 40 square centimeters (cm^2) (6.2 square inches (in^2)), a screen surface area greater than or equal to 903 cm^2 (140 in^2), and a hole size less than or equal to 1.59 millimeters (mm) (0.0625 inch (in)). This pocket design also provides the trash rack function because it prevents a single object from blocking a large portion of the screen.

Three separate screens are located inside the IRWST at the bottom of the tank. Two of the screens are located at opposite ends of the tank, each with a frontal area greater than 1.9 square meters (m^2) (20 square feet (ft^2)) and a surface area greater than 46.5 m^2 (500 ft^2). The third screen, with a frontal area greater than 3.7 m^2 (40 ft^2) and a surface area greater than 92.9 m^2 (1,000 ft^2), is located in the center of the tank. Each of the smaller screens supplies one of the two recirculation lines and is joined to the center screen through cross-connect piping, which distributes the flow. The lowest screening surfaces are located 0.2 m (6 in) above the IRWST floor to prevent debris from being swept along the floor into the screen. During recirculation, the only flow entering the IRWST is the steam condensate that forms inside of the containment shell and collects in the IRWST gutter. The gutter feeds the IRWST through two drainpipes that terminate within 1.5 m (5 ft) of the IRWST floor and at least 3.7 m (12 ft) away from any screen face, which prevents debris from entering the tank close to the screens. For minimum floodup conditions during recirculation, the water level is a few inches above the top of the screen assembly.

The containment recirculation sump for the AP1000 is the loop compartment. Two containment recirculation screens, each with a frontal area greater than 9.8 m^2 (105 ft^2) and a surface area greater than 232 m^2 (2,500 ft^2), are located next to one another along walls on the loop compartment floor. The loop compartment floor is 3.5 m (11.5 ft) above the reactor vessel cavity floor, which is the lowest level in containment. Each screen supplies one of the two recirculation lines. The screens are cross-connected with flow channels so that even if only one recirculation line is operating, both containment recirculation screens will be available to filter the flow. A 0.6 m (2 ft) high curb is located in front of the screens to prevent debris from being

swept along the floor into the screen. Protective plates, which cover the screen and extend outward at least 3 m (10 ft) in front of the screen face and 2.13 m (7 ft) beyond the sides of the screen, are located no more than 0.3 m (1 ft) above the top of the containment recirculation screens. These plates are designed to prevent debris from settling into the water close to the screens. During recirculation, even at minimum floodup conditions, the water level remains about 3 m (10 ft) above the top of the screens.

The AP1000 is equipped with an active nonsafety-related injection and recirculation system (the residual heat removal system or RNS). This system is an investment protection system and will be used following a LOCA if there is power and the system is available. The RNS initially injects water from the cask loading pit and then switches to recirculate the sump water. The system is not credited in the safety analysis; however, the applicant has evaluated the screens and core assuming the higher RNS flow rates to demonstrate that the RNS system will function following a LOCA.

During the AP1000 recirculation phase, some portions of the reactor coolant system (RCS) piping are submerged. LOCAs resulting from pipe breaks in these locations will then be flooded, allowing sump fluid to bypass the screens and flow directly to the core. Because of this, the core is evaluated as a separate debris filtering location.

6.2.1.8.2 Evaluation

6.2.1.8.2.1 Break Selection

A primary objective of break selection is to identify the break location that results in debris generation that produces the largest head loss across the screens. Section 3.3 of NEI 04-07 describes a process; whereby, different break locations are systematically evaluated to determine which is limiting. In the AP1000 design, there are only three sources of debris that transport with the recirculating water: latent or resident containment debris, debris from postaccident chemical effects, and debris from coatings located in the zone of influence (ZOI) of a LOCA jet. The limiting break for each filtering location (screens or core) is determined by examining the debris generation and transport assumptions for each of these debris types.

6.2.1.8.2.1.1 Containment Recirculation Screens and In-Containment Refueling Water Storage Tank Screens Break Selection

In the AP1000, the amounts of latent and chemical debris generated and transported to the containment recirculation and IRWST screens are independent of break location; therefore, the only consideration for break selection is coatings debris. Because all coatings in the ZOI are assumed to generate fine transportable particulates, the limiting break location is the one whose ZOI contains the largest amount of coatings. The applicant described its break selection process for cold leg (CL) and hot leg (HL) LOCAs in the June 30, 2010, Revision 2 response to RAI-SRP6.2.2-SPCV-25. The applicant determined that the limiting breaks would occur in the largest diameter lines: the main loop CL, which has a 55.9 centimeter (cm) (22 in) inner diameter (ID) and the main loop HL, which has a 78.7 cm (31 in) ID. These breaks encompass potential breaks on smaller lines because the main loop pipes are located in the same general area and have significantly larger ZOIs than the smaller lines.

The applicant then selected the terminal ends of the main loop lines as potential break locations and quantified the amount of both epoxy and inorganic zinc coatings at each site by multiplying the surface area of coated beams, pipes and flat surfaces located in the ZOI by the coating

thickness and density. The limiting CL location was the terminal end at reactor coolant pump (RCP) #2, in the west loop compartment. The limiting HL break was the terminal end at either steam generator, due to similarity between the compartments. The applicant then assessed potential break locations at 5 ft intervals along each CL and HL line, as recommended by the safety evaluation on NEI 04-07, to demonstrate that these sites, with potentially different coating inventories were bounded by the limiting breaks. This analysis employed plant layouts to identify the location of coated surfaces such as walls, plates, pipes and beams with respect to each postulated ZOI. When necessary, the applicant estimated the increase or decrease to each type of coated surface to show that the net amount of coatings was bounded by the limiting break.

The staff finds that the spectrum of breaks evaluated is acceptable because, while the unique characteristics of the AP1000 greatly simplify the break selection process, the applicant's procedure meets the intent of the NEI 04-07 and the related safety evaluation, and Regulatory Position C.1.3.2.3 of RG 1.82, Revision 3.

6.2.1.8.2.1.2 Core Break Selection

Because of the relatively high containment floodup level during long-term recirculation operation in the AP1000 design, some LOCA break locations will be flooded, resulting in a portion of recirculation flow with unfiltered containment debris entering the reactor core through the submerged break. The DECL break at the reactor vessel, DEHL break at the reactor vessel, and double-ended direct vessel injection (DEDVI) line break in the loop compartment are breaks that could be submerged in the flooded containment sump and allow unfiltered debris into the reactor vessel during the post-LOCA long-term cooling phase.

The objective of break selection is to choose a limiting LOCA break location for the GSI-191 long-term cooling evaluation. The break selection considers two aspects: (1) the limiting break, which contributes the most unfiltered debris to the core region causing core flow blockage; and (2) the limiting break for long-term core cooling evaluation, which results in the worst core heatup because of early containment recirculation initiation at a higher decay heat level.

In Section 6.2.1.8.2.6 of this report, the staff discusses the containment debris transport for the DECL break, DEDVI line break, and DEHL break and the evaluation of the percentage of the containment debris entering the reactor vessel for these breaks. The percentage of the debris that might be transported into the reactor vessel without screening by the containment recirculation screens is determined by integrating the relative recirculation flows through the break and through the intact DVI lines of the PXS. Flow split calculation for the DECL break identified that [ ] percent of the water in the containment will come through the DECL break, compared to the DEDVI line break with [ ] percent flow split. Therefore, the applicant determined that the DECL break at the reactor vessel is the limiting break with respect to debris transport to the core, with a flow split conservatively rounded up to 90 percent. The applicant assumed that all of the latent debris in the containment would be in the containment water and, therefore, 90 percent of containment debris would enter the reactor vessel through the submerged DECL break during the post-LOCA recirculation long-term core cooling phase, bypassing the IRWST and containment recirculation screens. This 90 percent debris bypass calculated with a DECL break is the design-basis value used to determine the containment debris bypass to the core for the AP1000 design.

For its long-term cooling evaluations performed in the AP1000 DCD, Revision 17, the applicant selected the DEDVI line break in the PXS room as the limiting long-term cooling case because it

minimized the cooling water injection head at the highest decay heat generation rate. The case analyzed in the DCD is the continuation of a small-break LOCA. The DCD long-term cooling evaluations used the PXS valve room as the break location since natural circulation flow losses from the break to the core inlet are greatest, thereby minimizing the amount of flow into the core inlet from the break. The DCD case did not consider the GSI-191 concern that evaluates debris entering the core through the break location. Since the DEDVI break in the PXS room has only a small amount of debris accessible to the break (i.e., only debris in the PXS room floodup water volume can enter the break location), the applicant considered the DEDVI line break in the loop compartment as the limiting break for unfiltered debris entry into the core.

The applicant performed long-term cooling sensitivity studies, described in APP-PXS-GLR-001, Revision 4. The objective of these sensitivity studies was to demonstrate that core cooling margins are maintained when large, arbitrary, nonmechanistic head losses are added to the containment recirculation screens, IRWST screens, and the core inlet. Section 6.2.1.8.2.7 of this report includes additional discussion of these sensitivity studies. As discussed in Section 4 of TR-26, the applicant selected the direct vessel injection (DVI) break for the long-term cooling sensitivity analyses with debris-induced core head loss for the following reasons:

- A DECL break has significantly less resistance to flow than the DVI break; therefore, the DVI break will minimize the water flow to the core. For this reason, only the DVI breaks were analyzed with WCOBRA/TRAC in APP-PXS-GLR-001, Revision 4.

- The lower elevation of the DEDVI break also allows water from the containment to flow into the RCS through the break sooner than for a DECL LOCA; therefore, the debris will start to accumulate sooner with a higher decay heat.

- A DEDVI break results in higher decay heat levels at the time recirculation begins compared to a DECL LOCA. With a DEDVI break, a portion of the IRWST injection flow will spill out of the break into containment, thereby reducing the time of IRWST injection.

In the post-LOCA long-term cooling sensitivity studies, described in APP-PXS-GLR-001, Revision 4, the DEDVI line break in a PXS room and in the loop compartment adjacent to the DVI inlet nozzle, respectively, were assumed from the analysis. In Cases 1 through 5 and Case 7, which have smaller assumed core inlet flow resistances, the DVI line break was assumed in the PXS room, where the floodup level is lower than the floodup level in the loop compartment and is, therefore, more conservative. In other cases with larger core inlet flow resistances simulating a larger amount of debris entering the core, the DVI break is assumed to be in the loop compartment. This is a more realistic assumption because a DVI break in the loop compartment would be exposed to all of the latent debris in the containment to the break location; whereas, a DVI break in the PXS room would only expose the break to the small amount of debris that is located in the PXS room. Therefore, significantly more debris is available to enter the DVI break in the loop compartment. The sensitivity study Case 10, which is a DVI break in the loop compartment with significant flow blockage at the core inlet, was determined to be the limiting break for water flow and debris to the core.

As discussed in Section 6.2.1.8.2.6 of this report, the HL break was determined not to be a significant deterrent for long-term cooling. Based on fuel assembly testing, the debris that enters the top of the core will be broken up by the two-phase flow leaving the core, and core cooling is maintained by the intact DVI flow.

On the basis of its review, the staff finds that the applicant's assumption that 90 percent of the total latent debris in the AP1000 containment can enter the core inlet through the DECL break is conservative, and that the applicant's determination that the DECL break is the design-basis break location for debris transport to the core and for potential debris blockage of the core inlet is acceptable. Additionally, the staff finds that the applicant's use of the DEDVI break in the loop compartment for the long-term core cooling sensitivity studies while using the 90 percent debris bypass obtained from the DECL break to determine the debris-induced core entrance head loss is conservative and acceptable.

6.2.1.8.2.2 Zone of Influence/Debris Generation and Characterization (Excluding Coatings)

The ZOI is the volume about the break in which the LOCA break jet forces would be sufficient to damage materials. Debris generation is the amount of debris generated by these forces, and debris characterization establishes the properties of this debris. The staff considered Section 3.4 of NEI 04-07 and the related safety evaluation when evaluating this section, which addresses all debris types except coatings, which are discussed in the following section.

In the AP1000, metal reflective insulation (MRI) or a suitable equivalent is specifically required on the reactor vessel, RCPs, steam generators, pressurizer, and all ASME Code Class 1 lines. MRI is also required at any location within the insulation ZOI, which is defined in DCD Tier 2, Section 6.3.2.2.7.1, for two situations:

(1) When there are intervening components, supports, structures, or other objects, the ZOI includes the spherical region within a distance equal to 29 IDs of the pipe break for Min-K, Koolphen-K or rigid cellular glass or 20 IDs of the pipe break for other types of insulation.

(2) When there are no intervening components, supports, structures, or other objects, the ZOI is a cylindrical volume extending out from the break a distance of 45 IDs along an axis that is a continuation of the pipe axis and a distance of 5 IDs radial to the pipe axis.

The first ZOI definition is consistent with Section 3.4 of the safety evaluation on NEI 04-07, which considers the spherical ZOI a practical approximation of the jet impingement damage zone and provides appropriate radial values for spherical ZOI for specific insulation types in SER Table 3-2. The spherical radii used to describe the AP1000 ZOI are bounded by the values in this table, except for rigid cellular glass, which NEI 04-07 and the safety evaluation do not address. Since no data are available for rigid cellular glass, the applicant used the maximum ZOI of 29 IDs from Table 3-2 of the safety evaluation on NEI 04-07. The staff accepts this approach, which is further supported by the statements in NUREG/CR-6808, "Knowledge Base for the Effect of Debris on Pressurized Water Reactor Emergency Core Cooling Sump Performance," issued February 2003, that this insulation will float indefinitely even if damaged. The second ZOI definition is identical to that certified in Revision 15 of the DCD. It was evaluated and found to be acceptable in NUREG-1793, Revision 0, based on existing tests and analysis.

If insulation in the AP1000 ZOI is not MRI, it must meet the DCD definition of suitable equivalence, which requires that the insulation be tested at conditions that bound the AP1000 operation and that if debris is generated it must not transported to any of the AP1000 filtering locations. It also requires that the NRC approve the test applicability and any subsequent analysis. This is appropriate because there are no clearly defined protocols for jet impingement testing, and all previous submittals on this type of testing were subject to staff evaluation.

DCD Tier 2, Section 6.3.2.2.7.1, Item 10 prohibits other potential sources of fibrous material, such as ventilation filters or fiber producing fire barrier, in the insulation ZOI. The staff agrees that this design commitment, in combination with the previously discussed insulation commitments, excludes all potential sources of fibrous debris except latent debris from the ZOI.

In RAI-SRP6.2.2-CIB1-28 (and supplements), the staff asked the applicant to clarify how concrete in containment is treated as a debris source, including chemical, coatings, and particulate debris. In letters dated January 10, 2010 and April 26, 2010, and July 30, 2010, the applicant explained that coatings are assumed to fail as particles within the ZOI and transport, and as chips outside the ZOI and not transport. In addition, the applicant explained that all concrete surfaces flooded following a LOCA are assumed to react with the water pool and potentially contribute to chemical debris. The staff found these assumptions acceptable, as discussed in Sections 6.2.1.8.2.3 and 6.2.1.8.2.4 of this report.

With respect to LOCA-generated particulate debris, the applicant explained that concrete is not a likely source of particulate debris for the AP1000 because the only concrete surfaces inside containment impacted by LOCA jets are the concrete floors. The containment walls and ceilings are constructed using steel-lined concrete modules. For AP1000, concrete debris is not considered in the composition of particulate debris based on the proximity of the concrete surfaces to the potential break locations, the orientation of the concrete with respect to the break, and the sharp decrease in the pressure of a LOCA jet as a function of distance.

In the July 30, 2010, letter, the applicant addressed the staff's question regarding the concrete damage apparent in the German HDR test results documented in NUREG/CR-0897, "Steam-Water Mixing and System Hydrodynamics Program, Quarterly Progress Report, Jan.-Mar. 1979". The applicant questioned the applicability of these results, since there is limited information about the testing and multiple tests led to the damage shown in NUREG/CR-0897. The applicant referenced WCAP-7391, "Pressurized Water and Steam Jet Effects on Concrete," as a more relevant study. This report documents results of jet impingement tests that were performed in 1970 by directing a [ ] pounds per square inch gauge (psig), water jet at [ ] concrete slabs. The July 30, 2010, letter also provides jet impingement pressures calculated for double-ended breaks in the limiting case of a pipe parallel to the AP1000 floor. As discussed below, the applicant performed further analysis using the WCAP-7391 test results and jet impingement calculations.

Although Section 1.3.2.4 of RG 1.82, Revision 3, states that erosion of concrete should be considered a potential source of particulate debris, the staff has not developed guidance for a ZOI or quantification of the debris. The applicant assessed the effect of potential concrete erosion debris on the available debris margins in the design basis. In its analysis, the applicant derived a threshold destruction pressure based on the WCAP-7391 jet impingement testing of concrete beams, and then determined whether any double-ended pipe breaks or longitudinal breaks ("split or side breaks") could exceed the threshold pressure. According to WCAP-7391, the tests did not erode the concrete itself, [ ]. In those tests, the minimum value of L/D (the length to diameter ratio which is the distance from the break to the target divided by the pipe inside diameter) was [ ] for the WCAP-7391 test conditions (CL conditions for the AP1000). The L/D ratio is a parameter used to characterize impingement pressure as a function of distance from a jet nozzle. Using the American National Standards Institute/American Nuclear Society (ANSI/ANS) 58.2-1988, "Design Basis for Protection of Light Water Nuclear Power Plants Against the Effects of Postulated Pipe Rupture," jet model to calculate pressure contours for

these conditions, the applicant calculated a stagnation pressure of [ ] at the concrete surface. For AP1000 HL conditions and a stagnation pressure of [ ], the corresponding L/D was [ ]. The applicant concluded that these tests demonstrated that jets from double-ended breaks in the AP1000 would not damage concrete at L/D values equal to or greater than [ ] for the HL and [ ] for the CL.

Therefore, for this margin assessment, the applicant set the acceptance criteria for concrete damage at [ ] and applied the same jet model to the double-ended break locations. From these calculations the applicant determined that no locations exceeded the [ ] criterion at the concrete surface, regardless of whether the pipe was oriented perpendicular or parallel to the floor. The staff found this approach acceptable because the WCAP-7391 tests showed no concrete damage and the staff has accepted the ANSI/ANS model as a basis for calculating pressure contours from LOCA jets. The staff evaluated the ANSI/ANS model in the December 2004, safety evaluation of NEI 04-07.

Continuing with the margin assessment, the applicant also evaluated the potential for concrete damage from longitudinal breaks (also referred to as "split breaks" or "side breaks") at the same locations (module floors). For the geometry of the break, the applicant used one-half of the pipe inside diameter as the width and two times the inside diameter as the length. The applicant used this approach to conform to Branch Technical Position (BTP) 3-4, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment." The applicant then used the ANSI/ANS 58.2-1988 jet model to determine the distance to where the impingement pressure is less than the [ ] acceptance criterion for concrete damage. Based on these calculated pressures for longitudinal breaks, the applicant identified five lines for which the calculated jet impingement pressure on concrete exceeds the [ ] acceptance criterion: [ ].

To estimate the amount of concrete debris generated by these breaks, the applicant assumed that all of the concrete becomes debris where the jet impingement pressure exceeds [ ]. [

]. The applicant also stated that the ellipsoid shape would be partially filled with steel reinforcement bar rather than concrete, but in this analysis the ellipsoid was assumed to be all concrete.

The applicant then assumed [

]. The applicant considered this a conservative size distribution relative to the appearance of the concrete damage in photographs of the German HDR test results documented in NUREG/CR-0897. The applicant did not attempt to estimate the size of fine debris in these photographs, but concluded that the concrete debris was mostly in the form of large pieces several inches in length and width. Given the lack of experience with concrete debris generation from impingement, the wide size range of the materials in concrete, and the appearance of the debris generated in the HDR test, the staff found that this size distribution is a reasonable assumption for the concrete debris. The entire concrete debris surface area was assumed to react chemically with the sump fluid.

By summing the concrete, coatings, and latent particle mass at each break location, the applicant concluded that all locations were bounded by the particulate quantities used in the screen and fuel assembly tests. With respect to chemical debris, all of the concrete particulate was assumed to react with the water, potentially generating additional sodium aluminum silicate and calcium phosphate. The applicant determined that the concrete particulate debris added approximately [ ] to the [ ] of concrete already used in chemical debris calculation, for a total of [ ]. (As indicated in Section 6.2.1.8.2.4 of the SER, all concrete in the post-LOCA flood-up zone was assumed, for the design basis analysis, to have the coating removed and, therefore, assumed to react chemically with the sump fluid.) The total calculated amount of chemical debris, including that from the concrete particulate, did not exceed 25.9 kilogram (kg) (57 pounds (lb)), which is the design basis amount of chemical debris that was scaled for use in the screen and fuel assembly testing. The staff performed confirmatory calculations and reached the same conclusion regarding the surface area of the concrete debris and the increase in chemical debris.

Based on this analysis the applicant concluded the following from the margin assessment: (1) the only concrete surfaces potentially exposed to impingement from a LOCA jet are limited floor areas of modules; (2) double-ended breaks of the pipes nearest the concrete do not generate concrete debris, regardless of orientation (perpendicular or parallel); (3) split or side breaks in the pipes nearest the concrete may generate concrete debris from erosion; and, (4) the estimated quantities of concrete particulate and associated chemical reaction debris are bounded by the screen and fuel assembly head loss tests that support the proposed licensing basis.

The staff found the applicant's evaluation acceptable to address Section 1.3.2.4 of RG 1.82, Revision 3, because the evaluation supports the position that concrete debris from jet impingement for the AP1000 plant is unlikely, and because a reasonable estimate of the amount of particulate and chemical debris that could be generated from concrete by a LOCA is within the AP1000 design basis.

Debris Characterization

The next step in the evaluation is to characterize the generated debris for input to the transport analysis. The applicant assumed that MRI in the ZOI degrades to pieces of crumpled foil as small as 1.3 cm by 1.3 cm (0.5 in by 0.5 in), which is consistent with the smallest classification of blast-tested MRI in NUREG/CR-6808. The applicant did not provide a quantity of degraded MRI because the subsequent transport analysis demonstrates that it will not transport to the screens or core.

6.2.1.8.2.3 Coatings

To determine if the AP1000 design meets the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5) with respect to protective coatings (paint) in containment, the staff reviewed the information in the DCD and supporting documents according to the guidance listed in Section 6.2.1.8 of this report. The following are key guidance documents for coatings debris, and the first two are exclusive to coatings debris:

- "Revised Guidance Regarding Coatings Zone of Influence for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,'" dated April 6, 2010 (ADAMS Accession Number ML100960495)

- Enclosure 2, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Coatings Evaluation," to "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,'" dated March 28, 2008 (ADAMS Accession Number ML080230234)

- NEI 04-07, Revision 0 dated December 2004, and the staff's accompanying safety evaluation

Sections 6.1.2 and 6.3.2 of DCD Revision 17 describe the selection and use of coatings for the AP1000. The three types of paint coatings in containment are epoxy, inorganic zinc, and unspecified manufacturer standard coatings on engineered components. The coatings are applied to three main types of surfaces inside containment: the inside surface of the containment shell, engineered components, and a variety of other surfaces. These other surfaces include both steel structures (walls, ceilings, floors, columns, beams, braces, and plates) and concrete structures (walls, ceilings, and floors). The plant design intentionally limits the amount of painted surface area within the ZOI of a LOCA jet.

The AP1000 design is similar to operating reactors with respect to coatings debris generation and transport. However, design features of the AP1000, such as the lack of containment spray during a LOCA and the low flow rates in the water pool are expected to reduce coatings debris generation and transport relative to operating reactors. For example, stainless steel is used to reduce the need for coatings. According to the descriptions in Section 6.3.2.2.7.3 of the DCD, stainless steel is used rather than coated carbon steel on surfaces within the coatings ZOI near the recirculation screens. The quantity of coatings is also reduced with respect to the originally certified AP1000 design by eliminating inorganic zinc as a primer for epoxy. Instead, inorganic zinc is used in the containment only on the inside surface of the containment shell and where the temperature during normal operating conditions exceeds the limit for epoxy.

The inorganic zinc coating inside containment is classified as safety Service Level I as defined in RG 1.54, Revision 1, which has guidance for procurement, application, inspection, and monitoring based on NRC-approved ASTM standards. The quality assurance program for these coatings meets the relevant requirements of Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50. Inside containment, self-priming, high-solids epoxy is used on steel structures (walls, ceilings, floors, columns, beams, braces, and plates) and concrete structures (walls, ceilings, and floors). Appendix B to 10 CFR Part 50 applies to the procurement of these coatings. As described in DCD Sections 6.1.2.1.6 and 6.1.3.2, COL applicants are responsible for the programs to address quality assurance. Section 6.1.3.2 requires COL applicants to provide a program to control the procurement, application, inspection, and monitoring of Service Level I and III coatings, as well as most Service Level II coatings inside containment. The following paragraphs discuss these requirements in more detail below.

The ZOI for coatings conforms to the most recent (April 6, 2010) staff guidance on coatings evaluations for resolution of GSI-191. Specifically, the ZOIs are four times the pipe diameter (4D) for epoxy coatings and 10 times the pipe diameter (10D) for inorganic zinc coatings (DCD Section 6.3.2.2.7.1). The applicant determined the quantity of epoxy and zinc coatings by estimating the surface area with the limiting HL and CL ZOIs, and then applying the thickness and density in the plant specifications already developed. The methodology for selecting pipe break locations to calculate debris quantities is described in Section 6.2.1.8.2.1.1 of this report.

Briefly, the limiting CL location was the terminal end at RCP #2, in the west loop compartment. The limiting HL break was the terminal end at either steam generator, due to similarity between the compartments.

Using plant layout information, the applicant then identified the location of coated surfaces, such as walls, plates, pipes and beams with respect to each postulated ZOI. As explained in its June 30, 2010, Revision 2, response to RAI-SRP6.2.2-SPCV-25, the applicant calculated epoxy coating debris quantities of [ ] and [ ], respectively, for the limiting CL and HL breaks. For these locations, no inorganic zinc coating was located within the 10D ZOI. The staff performed confirmatory calculations based on the applicant's surface area, density, and thickness values. The staff concluded the assumed thickness value (2 coats at 0.01524 cm (0.006 in) each) was conservative for calculating coating weight because it matched the high end of the recommended thickness for nuclear-grade epoxy coatings. The staff also concluded the density value ([ ]) was conservative for calculating coating weight because it was higher than the minimum value (1602 kg/m^3 (100 lb/ft^3)) required by the design.

The applicant's analysis of coating debris quantity is based, in part, on a design change from coated carbon steel to uncoated stainless steel for ASME Code Section III steam generator system instrument piping. This change was described in the June 30, 2010, response, to RAI-SRP6.2.2-SPCV-25 and was submitted to the NRC in a letter dated July 8, 2010. The staff's evaluation of this proposed piping change is described in Section 23.P of this report. The assumed form of the coating debris also conforms to the staff guidance on coatings discussed above. Outside the ZOI, epoxy coatings are assumed to fail as chips and not transport to the screens, as discussed below. Inorganic zinc outside the ZOI is assumed to remain intact because of its Service Level I designation, which conforms to the staff guidance for qualified coatings outside the ZOI. Coatings within the ZOI are assumed to fail in the form of fine particles that transport both to the screens and core. The staff believes it is necessary to treat the coatings in the ZOI as fine particles for two reasons. First, coating chips are assumed not to transport, as discussed below. Therefore, only fine particles of coating would contribute to the debris loading at the screens and fuel assemblies. Second, as explained in the staff's March 28, 2008, guidance for coatings debris, it is important to treat coating debris from the ZOI as particles if there is a possibility of a thin, filtering bed. Since fuel assembly testing indicated head loss resulting from a filtering bed, it was appropriate to consider coating debris as fine particles that transport.

The assumption that Service Level II epoxy outside the ZOI will fail in the form of chips conforms to the guidance because the coatings will be treated as degraded qualified coatings. These coatings are procured (with one exception, as noted in DCD Section 6.1.2.1.6) under Appendix B to 10 CFR Part 50 and are tested under DBA conditions. However, the Service Level II coatings will not have the same level of quality assurance and assessment requirements as fully qualified coatings in containment (i.e., Service Level I). The staff determined that it is appropriate to treat this combination of product qualification and subsequent quality assurance as a degraded qualified coating for debris analysis and assume these coatings in accordance with the staff's March 2008 guidance. The exception to the procurement qualification is for the epoxy coatings in the CVS room that connects to the containment only through a drain line that discharges to the waste processing system below and away from the recirculation screens. As explained in DCD Section 6.1.2.1.6 and Table 6.1-2, the epoxy coatings in this room are not required to be procured under Appendix B to 10 CFR Part 50.

Debris in the form of chips is assumed not to transport to the screens because of the coating's high density, which is a design requirement. Design features of the AP1000 are also expected to keep debris from entering the pool near the screens. Coatings are required to have a density of at least 1.6 grams per cubic centimeter (g/cm^3) (100 lb/ft^3) (Epoxy will be purchased as DBA qualified. Epoxy outside the ZOI is treated according to the Keeler & Long tests for operating reactors and assumed to fail as chips. This testing (Keeler & Long Report No. 06-0413) was performed in support of GSI-191 resolution for operating plants and simulated debris generation from DBA-qualified epoxy and inorganic zinc coatings. The material tested was in the form of epoxy chips with an attached inorganic zinc primer. Nearly all of the epoxy remained as chips larger than 0.79 millimeter (mm) (1/32 in) in diameter, while the inorganic zinc failed as particulate and disbonded from the epoxy. The NRC guidance noted above recommends using these data in conjunction with coating chip transport data to reduce the amount of degraded qualified coatings assumed to transport to the screens.

The staff guidance dated March 28, 2008, states that if less than 100 percent transport of the coatings debris is considered, the basis for the debris settlement should be provided, such as the NRC-sponsored coating chip transport testing described in NUREG/CR-6916, "Hydraulic Transport of Coating Debris," issued December 2006. This testing found that at a steady-state velocity of 0.06096 meters per second (m/s) (0.2 feet per second (ft/s)), most epoxy coating chips with a density (2002 kg/m^3 (125 lb/ft^3)) similar to the specified density for AP1000 coatings ([ ]) did not transport to the end of the test flume at a water velocity of 0.06096 m/s (0.2 ft/s). The calculated maximum approach velocity range for the AP1000 is less than [ ] even for the most limiting case, which occurs [

]. Therefore, the test conditions in NUREG/CR-6916 bound the AP1000 flows.

With respect to density, Service Level II coatings on structures in the AP1000 containment (except in the CVS room described above) and on engineered components in defined areas are required to have a density of 1602 kg/m^3 (100 lb/ft^3) in order to take credit for settling. The defined areas for engineered components are locations below the maximum flood level of a design-basis LOCA or above the maximum flood level and not inside a cabinet or enclosure. This requirement appears in DCD Tier 1, Table 2.2.3-4. As explained above and in DCD Section 6.3.2.2.7.1, containment recirculation screens have a debris curb as well as protective plates that extend at least 3.04 m (10 ft) in front and 2.13 m (7 ft) to the side of the recirculation screens. Considering these design features, the low fluid approach velocities at the screens, the specified coating density, and the NUREG/CR-6916 test data, the staff finds it reasonable to assume for the AP1000 plant design that coating chips (generated outside the ZOI) will not transport to the containment recirculation screens.

Similarly, for the IRWST, since wetted surfaces inside the IRWST are made of corrosion-resistant materials (e.g., stainless steel); no paint coatings are used on surfaces near the IRWST screens. As described in DCD Section 6.3.2.2.7.2, the IRWST is covered during operation, and the bottom of each vertically oriented screen is 15.24 cm (6 in) above the floor. Therefore, the only route to the IRWST for coatings debris is through the gutter system. However, because of the location of the gutter, the gutter trash rack, and the gutter discharge piping and the high density of the coatings, it is reasonable to assume that coating debris chips will not transport to the IRWST screens. Therefore, based on chip density, transport distance, and water velocity, the staff finds it acceptable to assume for the AP1000 plant that the epoxy chips will not transport to the containment recirculation or IRWST screens.

The staff's evaluation included review of new head loss testing for the recirculation screens, IRWST screens, and fuel assemblies. The testing performed for the screens and fuel assemblies included the amount of ZOI coatings. These tests used silicon carbide particles, with an average size of [ ], as the surrogate for coatings debris. As stated in the safety evaluation for NEI 04-07, the staff found it reasonable to treat coating debris as highly transportable particulates in the size range 10 to 50 microns where there is a possibility of a thin fiber bed. This particle size is based on the basic material constituents for inorganic zinc and epoxy coatings. Section 6.2.1.8.2.8 of this report discusses the test program and results.

For the reasons discussed above, the applicant's assessment using the NRC guidance in RG 1.82, Revision 3, for protective coatings is acceptable with respect to the ZOI, quantity and form of coating debris assumed, and the representation of coatings debris in the screen and fuel assembly testing. On this basis, the staff concludes that the AP1000 plant design meets the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5), as they relate to the effect of protective coatings debris on long-term cooling following a LOCA, including specific consideration of the effects of protective coating debris under accident conditions.

6.2.1.8.2.4 Chemical Effects

To determine the compliance of the AP1000 design with the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5) with respect to chemical debris formed in the post-LOCA containment pool, the staff reviewed the information in the DCD and supporting documents using the guidance listed in Section 6.2.1.8 of this report. The following are the key guidance documents for chemical debris, and the first two are exclusive to chemical debris:

- Enclosure 3, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations," to "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,'" dated March 28, 2008 (ADAMS Accession Number ML080230234)

- "Final Safety Evaluation by the Office of Nuclear Reactor Regulation: TR WCAP-16530-NP, 'Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids To Support GSI-191,'" dated December 21, 2007 (ADAMS Accession Number ML073521294)

- NEI 04-07, Revision 0 and the staff's accompanying safety evaluation.

The staff's review of the applicant's supporting documents included an audit of detailed analyses and a calculation note documenting that applicant determined the chemical debris constituents and quantities using APP-PXS-M3C-052, "AP1000 GSI-191 Chemistry Effects Evaluation," Revision 1, dated May 4, 2009.

The applicant calculated debris from chemical precipitation using the methodology in WCAP-16530-NP. This methodology was developed for operating pressurized-water reactors (PWRs), and the safety evaluation listed above documents the staff's evaluation of the methodology for operating reactors. The staff's review of the methodology finds that the methods of WCAP-16530-NP also apply to the AP1000 based on the containment materials, the sump pH transient, and the pH buffering agent (trisodium phosphate (TSP)). With respect to temperature, however, the applicant noted in Section 2.4 of TR-26 that the predicted sump pool

temperature for the AP1000 is briefly outside the range evaluated for WCAP-16530-NP (60 °C to 132.2 °C (140 °F to 270 °F)) but concluded that the methodology was still applicable. The staff reviewed the amount and duration of the temperature deviation during the audit of the calculation note referenced above. The staff estimated that the temperature was between [ ]. The staff concluded that it was appropriate to apply the release rate equations in WCAP-16530 during this period because of the short time relative to the 30-day event and because corrosion data for [ ] did not indicate a sharp increase in the corrosion rate in this temperature range (see Reference 6.2-3 in WCAP-16530-NP).

The amount of chemical precipitation predicted by the applicant was small relative to typical operating plants because of the AP1000's design features. The lack of fiberglass insulation eliminates a potential source of dissolved silica that could precipitate as silicate compounds. Because the AP1000 contains no calcium silicate insulation, there is no precipitation of calcium phosphate resulting from interaction of calcium silicate and the TSP pH buffer. For operating reactors with TSP buffer, calcium phosphate is one of the key chemical effects. The AP1000 analysis assumed calcium phosphate would be generated based on interaction between the dissolved TSP and calcium dissolved from the concrete. The quantity of $Ca_3(PO_4)_2$ precipitate was determined by first calculating the amount of concrete dissolution using the methodology in WCAP-16530-NP and then assuming all of the dissolved calcium precipitated as $Ca_3(PO_4)_2$. In its safety evaluation of WCAP-16530-NP, the staff concluded that this approach is conservative.

Testing and analysis for operating PWRs have shown that aluminum corrosion is a key contributor to chemical debris in the presence of an alkaline water pool. For the AP1000, the only intended use of aluminum is the ex-core detector housings, and the AP1000 design specifies stainless steel or titanium covers to isolate the aluminum from the water. The design includes an ITAAC to verify that the detectors are enclosed in stainless steel or titanium (DCD Tier 1, Table 2.2.3-4, Item 8.c). Although these covers eliminate the potential chemical interaction between aluminum and alkaline water during accident conditions, the applicant anticipated that some quantity of aluminum interacting with the water may be unavoidable in certain engineered components. Therefore, the applicant set an upper limit for this aluminum (27 kg (60 lbs)) in the design (DCD Section 6.1.1.4) and evaluated this quantity for aluminum corrosion and precipitation using the WCAP-16530-NP methodology. In its May 13, 2009, response to RAI-SRP6.2.2-CIB1-21, the applicant stated that the actual quantity of this aluminum would be tracked in accordance with a design calculation note. The staff finds the approach acceptable because the certification documentation in Tier 1 and Tier 2 of the AP1000 DCD limits the amount of wetted aluminum to the amount analyzed for chemical debris.

The applicant's analysis produced the following calculated mass of chemical debris:

Aluminum oxyhydroxide (AlOOH)	[ ]
Sodium aluminum silicate ($NaAlSi_3O_8$)	[ ]
Calcium phosphate ($Ca(PO_4)_2$)	[ ]
TOTAL	[ ]

In the WCAP methodology, these precipitates formed from aluminum released from metallic aluminum and concrete, silicon released from silica powder (Min-K insulation) and concrete, and calcium released from concrete. With respect to the total amount of precipitate, WCAP-16530-NP assumes all dissolved calcium, in the presence of phosphate, and all dissolved aluminum form precipitates. In its safety evaluation on WCAP-16530-NP, the staff found that this is a reasonable assumption for calcium and a conservative assumption for aluminum.

The staff reviewed the basis for the release of chemicals and precipitation of debris during its May 7, 2010, audit of APP-PXS-M3C-052, Revision 1. For example, the applicant assumed the coating was removed from all concrete in the flooded zone, and the WCAP release rate equation included the corresponding concrete surface area. For insulation in which silicon is released from silica powder during accident conditions (i.e., Min-K insulation), the release equation assumed that [ ] of the insulation was washed into the pool and subject to chemical release. The staff finds this is a reasonable assumption, since non-reflective metallic insulation (RMI) insulation in containment must be enclosed in seal-welded stainless steel and qualified through NRC-approved testing to show that a LOCA would not generate chemical debris (see DCD Section 6.3.2.2.7.1).

For aluminum, the applicant assumed a surface area of [ ]. The staff considered this a reasonable assumption because it corresponds to a relatively thin sheet of aluminum, on the order of 0.25 cm (0.1 in). This results in a relatively high calculated aluminum release because the release rate is proportional to surface area, and lower thickness corresponds to higher surface area for a given mass. In addition, it is the same conversion factor used in the spreadsheet in Appendix D to WCAP-16530 for calculating the mass of aluminum released. The staff used these assumptions, along with the temperature and pH profile provided in the calculation note, to perform independent confirmatory calculations using the WCAP-16530 spreadsheet. The staff also performed the calculations manually using the release rate equations derived in WCAP-16530-NP and incorporated into the spreadsheet. The staff's calculations produced the same values as the applicant's.

The AP1000 analysis assumes that all chemical debris transports (see Table 3-4 in TR-26). Therefore, the screen and fuel assembly testing included the entire chemical debris load, represented by AlOOH surrogate. The AlOOH precipitate is [ ] of the calculated mass of chemical debris for the AP1000. According to the test reports for screen and fuel assembly testing (WCAP-16914 and WCAP-17028, respectively), the surrogate chemical debris was prepared and added according the WCAP-16530-NP procedures. The staff's safety evaluation on WCAP-16530-NP states that surrogate precipitate prepared in accordance with the directions on WCAP-16530-NP provides adequate settlement and filterability characteristics to represent post-LOCA chemical precipitates in strainer head loss tests. The AlOOH precipitate was generated outside the test loops for the design-basis tests. In some supplemental (engineering) tests, the raw chemical components were added to the loop, and the precipitates formed in situ. The debris was added both sequentially, according to the WCAP, and coincidentally, to address whichever case is more limiting for the AP1000. The quantity of debris is intended to represent the generation and transport rate in the plant and meets the staff's March 28, 2008, guidance on chemical effects.

For the reasons discussed above, the staff concluded that the applicant's assessment of chemical effects conforms to the staff's March 2008 guidance with respect to the type and quantity of chemical debris, as well as the representation of chemical debris in the screen and fuel assembly testing. Therefore, the staff concludes that the AP1000 plant design meets the requirements of GDC 35, GDC 38, and 10 CFR 50.46(b)(5), as they relate to the effect of chemical debris on long-term cooling following a LOCA, including specific consideration of the effects of chemical interaction and formation of debris under accident conditions.

6.2.1.8.2.5 Latent Debris

Latent or resident debris is dirt, dust, lint, and other miscellaneous material that is present inside containment during operation. As stated in DCD Tier 2, Section 6.3.2.2.7.1, Item 12, the design basis for the total amount of resident debris inside the AP1000 containment is 59.0 kg (130 pounds-mass (lbm)), of which up to 3.0 kg (6.6 lbm) is fiber. Additionally, COL Information Item 6.3-1, discussed in DCD Tier 2, Section 6.3.8.1, requires that COL applicants referencing the AP1000 design commit to a cleanliness program that limits the latent debris inside containment to these same quantities.

In TR-26, the applicant explained that limiting the total latent debris inside containment to 59.0 kg (130 lbm) is consistent with the practice for operating reactors. The information regarding latent debris in operating reactors came from publically available responses by individual plants to generic letter (GL) 2004-02, "Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors," dated September 13, 2004. The plants generally identified two quantities for latent debris: a walkdown value representing an estimate of the debris inside the plant based on physical sampling of containment surfaces, and an analysis (or bounding) value used in the GL 2004-02 evaluation. In accordance with Table 2-1 of TR-26, the average amount of latent debris in the 34 cited plants was 40.8 kg (90 lbm) based on walkdown data and 73.5 kg (162 lbm) based on the bounding values. The applicant attempted to correlate the amount of latent debris to the dominant type of insulation used in the plant (RMI or fibrous) and to the plant size, but it determined that neither was a strong predictor for the quantity of latent debris. The applicant concluded that other factors, such as the utilities cleanliness program, would be more indicative of the amount of debris found in the plant but did not provide additional data.

The staff's review of TR-26, Table 2-1, found that while the values were consistent with the individual plant responses to GL 2004-02, the table only included data from half of the 69 operating PWRs. Therefore, while the reported averages are not representative of the entire PWR fleet, the data demonstrate that some operating PWRs do have latent debris bounding values of less than 59.0 kg (130 lbm). The staff emphasizes the bounding values over the walkdown values because, while some plants performed rigorous walkdowns in accordance with the guidance in the NEI 04-07 safety evaluation, other plants recognized that their walkdown calculations were not bounding; therefore, margin was added to the results before performing the analysis.

Rather than defining the fiber quantity as a mass percentage of total latent debris, the applicant has set the design basis for fiber inside the AP1000 containment to 3.0 kg (6.6 lbm), which is 5.1 percent of the total latent debris. In TR-26, the applicant explained that this is consistent with data from Table 2 of NUREG/CR-6877, "Characterization and Head loss Testing of Latent Debris from Pressurized-Water-Reactor Containment Buildings," issued July 2005, which shows that the percentage of fiber in two of the four sample plants is less than 4 percent. The NEI 04-07 safety evaluation referenced this same NUREG to support the recommendation to treat 15 percent of the total latent debris mass as fiber. The different conclusions result from two different treatments of the NUREG-reported fiber-to-particulate compositions. In the data referenced by the NEI 04-07 safety evaluation, objects larger than a 0.335-cm (0.132-in) mesh sieve were removed from the sample because they were assumed to be nontransportable. In the data referenced in TR-26, the assumed nontransportable objects were retained because the applicant assumed that all latent debris is transportable and did not anticipate that utilities will remove the larger pieces of debris during sampling. While these assertions seem reasonable, the staff finds the design-basis quantities of total latent debris and fibrous latent debris inside the AP1000 containment to be acceptable because of the COL commitment to limit latent debris to the design-basis amounts. In accordance with the containment cleanliness program,

applicants referencing the AP1000 must include a program to limit the amount of latent debris left inside containment following refueling and maintenance outages to 59.0 kg (130 lbm) total latent debris, of which up to 3.0 kg (6.6 lbm) is composed of fibrous material.

As described in Section 3.5.2.2.2 of NEI 04-07 and its related safety evaluation, miscellaneous debris such as equipment tags, tape, and stickers or placards affixed by adhesives that could become transportable should be considered part of latent debris. In accordance with the NEI 04-07 safety evaluation, if the lanyards or adhesives fail and the signs are transported to the screen intact, the available screen surface area should be reduced by an appropriate amount, which is called the sacrificial screen area. No sacrificial screen area is used in the AP1000 evaluation because the DCD requires that all potential sources of transportable material (such as caulking, signs, and equipment tags) be designed so they do not produce debris that will be transported to the filtering areas.

Specifically, unless the items are located inside a cabinet or enclosure, either they must be high density or testing must be performed at conditions that bound the AP1000 to demonstrate that the debris do not transport to any AP1000 filtering location or generate chemical debris. If testing is performed, the NRC must approve it.

6.2.1.8.2.6 Debris Transport

Debris transport is the estimation of the fraction of debris that is transported to each of the filtering locations (screens or core), considering blowdown transport, washdown transport, pool fillup transport, and recirculation transport. In the AP1000 design, washdown transport does not come from containment sprays but from PCS operation, where the majority of the condensed steam runs down the containment walls and the remainder falls from the containment dome. Debris transport is discussed in Section 3 of TR-26 and is evaluated as follows considering Section 3.6 of NEI 04-07 and the related safety evaluation.

All coatings in the ZOI are assumed to fail as fine particles and transport to the screens and core. Coatings outside the ZOI are assumed to remain intact or fail as chips that do not transport because of the high density. Section 6.2.1.8.2.3 describes coatings debris in more detail.

All of the chemical debris is assumed to transport to the screens and core. Section 6.2.1.8.2.4 describes the chemical debris in more detail.

The AP1000 analysis conservatively assumes that 100 percent of the latent debris in the containment will be transported by the fluid streams to the recirculating pool and remain entrained until it reaches the containment recirculation screens. The analysis assumes that 50 percent of the fibrous portion of the latent debris will transport to the IRWST screens. The applicant stated that this is conservative because the only path for debris to enter the closed IRWST tank is through the gutters, located just below the operating deck elevation along the containment walls. Debris transport from the operating deck to the gutter is limited by a border plate around the edge of the operating deck and a physical gap between this border plate and the gutter. Any fluid that makes it past the border plate but not across the gap will drain to the sump. Additionally, the operating deck is flat and has other unobstructed openings where water can spill directly into the containment sump. Because much of the latent debris in the AP1000 containment is located below the operating deck, it is not physically possible for it to be washed into the IRWST gutters. The staff agrees that the assumption of 50 percent transport of fibrous latent debris to the IRWST is conservative based on the IRWST gutter and operating deck

configuration. During RNS operation in the recirculation mode, it is possible that some fluid will backflow through the containment recirculation screen to the IRWST. It is expected that the containment recirculation screen will capture the fibrous portion of the debris as it backflows, but not necessarily the particulates. To conservatively account for this phenomenon, the applicant assumed that the IRWST screens see 100 percent of the particulate portion of latent debris.

In the AP1000 design, MRI debris is not transported to the screens or core because of the low velocities of the recirculating water. Based on testing described in NUREG/CR-6808, a velocity of at least 0.061 m/s or 3660 liters per minute per square meter (Lpm/m^2) (0.2 ft/s or 90 gpm/ft^2) is required to move a settled piece of crumpled MRI debris that is 1.3 cm by 1.3 cm (0.5 in by 0.5 in). The maximum fluid approach velocities at the containment recirculation and IRWST screens that occur during RNS operation and as shown in Table 5-2 of WCAP-15914-P are significantly less than this value. The flow through the containment recirculation corridor, which is identified as the limiting area approaching the containment recirculation screen in the Revision 1 response to RAI-SRP6.2.2-SPCV-13, is also below this value.

In order to limit debris introduced during pool fillup, the AP1000 design restricts fibrous material in the containment outside the ZOI but below the maximum floodup level. Insulation located here must be MRI, jacketed fiberglass, or a suitable equivalent. Also, other potential sources of fibrous material such as ventilation filters or fiber-producing fire barriers are not permitted below the floodup level. The MRI and jacketed fiberglass insulation will not be dislodged or eroded by the recirculation flow rates and, thus, will not generate debris. A suitable equivalent insulation is defined as one that has been tested at conditions that bound the AP1000 operation to demonstrate that recirculation flows will not generate chemical debris, and if any other type of debris is generated, it must not be transportable. The NRC must approve test applicability and any subsequent analysis.

Other potential sources of nonfibrous transportable material in containment, such as caulking, signs, and equipment tags, are required to be either made of high-density material, located in an enclosure, or demonstrated to not transport during testing. If they are made of high-density material, which is defined as greater than 1.6 g/cm^3 (100 pounds per cubic foot (lb/ft^3)), it is expected that they will settle and not be carried by the low AP1000 velocities. The minimum 2.5-hour delay between the accident and start of recirculation provides a reasonable amount of time for debris to settle on the containment floor.

While the applicant considered washdown from condensed steam running down the containment walls, it did not consider washdown associated with the containment spray system, which could transport more debris into the fluid stream. The staff agrees that it is not necessary to consider the impact of the containment spray system because it is a nonsafety-related system that will only be used in the event of a severe accident. During a severe accident, core heat removal or coolant has already been lost, and the containment spray's effect in transporting additional debris is not significant. Also, inadvertent actuation of the containment spray system during power operations has previously been found to be noncredible, as described on pages 6-25 of NUREG-1793.

For the AP1000, some break locations will be submerged once the containment has flooded up during the post-LOCA recirculation core cooling phase. When the containment water level is at the elevation of the break, in addition to the flow through the PXS intact DVI recirculation lines, flow carrying unfiltered debris can enter directly into the reactor vessel through the submerged break. A DECL break or DEHL break at the reactor vessel and a DEDVI line break can be submerged when the final containment floodup water level is reached. Any one of these breaks

could allow direct transport of unfiltered debris into the reactor, bypassing the containment recirculation screens, during the long-term cooling phase. The DECL break and DEDVI break could allow unfiltered debris into the downcomer, lower plenum, and ultimately into the core, potentially resulting in blocking reactor coolant flow through the core, and the DEHL break allows unfiltered debris into the upper part of the core area.

Of the DEDVI and DECL breaks, the DECL break potentially provides the greatest amount of unfiltered debris to the core inlet and the highest percentage of debris bypass (i.e., the percentage of the debris that could be transported into the RCS without filtering by the IRWST and containment recirculation screens). The applicant determined the percent of debris bypass for the DEDVI break and DECL break by calculating the flow split between the recirculation flows through the break and through the PXS intact DVI lines. Section 3.3 of TR-26 discusses debris transport to the core and calculations to determine the flow split (i.e., the percentage of recirculation flow entering the reactor vessel through the break carrying unfiltered debris). For the flow split calculation, the applicant conservatively assumed that [ ] would transport the entire available debris load to the recirculation screens and into the reactor vessel through the break and ultimately to the core. Therefore, the flow split is determined by integrating the relative recirculation flows through the break and through the PXS for [ ] at the containment floodup water level elevation. Flow rates through the break and through the PXS were obtained from WCOBRA/TRAC long-term cooling cases. The percentage of the water mass that has entered through the flooded break and through the PXS is determined after [ ]. The applicant performed the calculation for a DEDVI LOCA and for a DECL LOCA at the reactor vessel nozzle. In RAI-SRP6.2.2-SRSB-41, the staff asked that the applicant clarify the flow split discussion for the DECL and DEDVI breaks and related calculation presented in TR-26. The applicant responded by revising TR-26 to provide more information concerning the DEDVI and DECL flow splits and break calculations in Tables 3-1 and 3-2, respectively.

For a DECL break, the flow split would be greater than 10 percent flow through the PXS recirculation flowpath and [ ] percent flow through the rupture in the CL, which is greater than [ ] percent for the DEDVI break. The applicant assumed that: (1) 90 percent of the containment water would flow through the break in the CL; and (2) 90 percent of the fiber debris and 100 percent of the particulate debris in the containment would directly enter the reactor core through this DECL break. The applicant used the 90 percent unfiltered fibrous debris assumption in its fuel assembly testing, documented in WCAP-17028-P, by estimating a total fiber load in the containment of 2.99 kg (6.6 lbs) and multiplying that amount by 90 percent, yielding an estimated 2.69 kg (5.94 lb) of fiber directly entering the reactor core.

The staff also evaluated the debris transport through a HL break. In RAI-SRP6.2.2-SRSB-31, the staff asked the applicant to clarify the reasons why the DEHL break is not the most limiting break with respect to debris transport. Through testing and thermal hydraulic evaluations, The applicant developed the probable scenario that occurs during a DEHL break. During a postulated HL break, containment floodup water carrying unfiltered debris begins to enter the top of the core region through the submerged break, and water filtered by the IRWST and recirculation screens also flows into the downcomer and into the reactor core inlet through the two intact DVI lines. The debris that enters the upper core region through the break could block the upper core area, and some could be discharged through the automatic depressurization system (ADS) Stage 4 (ADS-4) valves connected to the HLs. The debris that is not discharged from the RCS through the ADS-4 valves can enter the upper portion of the core and flow down though peripheral low-power fuel assemblies. In its evaluation, the applicant identified that at

some point, because of the cross-flow thermal hydraulics in the AP1000, the downflow of water through the peripheral fuel assemblies will cross into the hotter fluid that is rising through the core, and the debris carried by the water that entered the break will mix with the other filtered water flowing into the core. As a result, some of the debris from the HL break could be captured on the upper tie plates of the core.

In its response to RAI-SRP6.2.2-SRSB-31, the applicant further reported that it conducted fuel assembly debris-loading head loss testing to estimate the amount and impact of debris that could be captured in the upper core region. WCAP-17028-P, designating the tests evaluating HL flow conditions as tests 35, 38, and 39. The applicant demonstrated through the fuel assembly testing that on the top of the fuel assembly debris beds will not form in the presence of two-phase flow, which would be prevalent in the AP1000 core during long-term cooling. As a result of this test, the applicant concluded that fibrous debris that enters through a HL break would ultimately be purged and captured by the IRWST and containment recirculation screens, thus purging the upper core region of debris over time during long-term cooling. The applicant further stated that debris that breaks loose from the top of a fuel assembly would likely discharge through ADS-4 valves and subsequently be filtered by the IRWST and containment recirculation screens. Section 6.2.1.8.2.7 of this report presents more discussion of these breaks.

On the basis of its review of the RAI-SRP6.2.2-SRSB-31 response and the associated fuel assembly testing in WCAP-17028-P, the staff finds that debris plugging in the core from a DEHL break is not the most limiting break with respect to debris transport and accumulation.

In summary, in Table 3-4 of TR-26, Revision 8, the applicant presented the AP1000 licensing basis fibrous, particulate, and chemical debris that could be transported into the reactor vessel and potentially reach the fuel assemblies. These debris loads were based on a total latent debris load of 58.96 kg (130 lbs), a total ZOI coating fine particles of 31.75 kg (70 lbs), and a chemical debris load of 25.85 kg (57 lbs). Of the 58.96 kg (130 lbs) of latent debris, no more than 2.99 kg (6.6 lbs) was estimated to be fibrous.

In TR-26, the applicant reported that chemical debris load is based on the type and quantity of chemical precipitates that may form in the post-LOCA recirculation fluid for the AP1000 design. The evaluation presented above identified the DECL break as the worst case break that provides debris to the core.

Based on the DECL break flow split calculation performed by the applicant, 90 percent of the water that enters the reactor vessel during long-term cooling comes through the CL break and 10 percent comes into the reactor as filtered water through the intact DVI lines. For the purpose of these evaluations, the applicant assumed that 90 percent of all fiber and 100 percent of all particulates and chemicals in the containment are transported to the reactor unfiltered.

On the basis of its review of debris transport analysis to the core, the staff finds that the applicant has conservatively evaluated the quantity of debris transported and the minimum transport time to the reactor core.

6.2.1.8.2.7 Availability of Long-Term Core Cooling

The AP1000 PXS and containment system are designed to continuously provide adequate cooling of the reactor following a LOCA. After the initial injection of water from the core makeup tanks (CMTs) and the accumulators, the IRWST provides safety injection by gravity drain. As

the IRWST injection continues, the containment sump water floods up above the reactor vessel. After the IRWST level drops to a low setpoint, the containment recirculation valves are open to provide the RCS with recirculation water from the containment sump to maintain core cooling indefinitely. During the long-term cooling phase of the LOCA, steam released from the reactor through the ADS-4 valves is condensed on the inner surface of the steel containment vessel, which is cooled on the outside by the PCS. The condensed water in the containment is collected in a gutter and returned to the IRWST. Both the IRWST and containment recirculation have screens to filter debris and protect the PXS flowpaths into the RCS.

Subsequent to a LOCA that occurs with the break below the post-LOCA water level in the containment, significant unfiltered debris can enter the reactor vessel through the submerged break location.

DCD Tier 2, Section 15.6.5.4C, describes the long-term cooling analysis to demonstrate that the passive safety systems provide adequate emergency core cooling system (ECCS) performance during the IRWST injection and containment recirculation duration. The analysis was performed for a DEDVI line break in the PXS room using the WCOBRA/TRAC code during the IRWST injection phase continuing into containment sump recirculation. When a quasi-steady state was achieved, a WCOBRA/TRAC window mode of calculation was performed. The boundary conditions for the WCOBRA/TRAC analysis, such as the containment pressure and the level and temperatures of the liquid in the containment sump, are based on the WGOTHIC calculations. The long-term cooling analysis demonstrated that: (1) the core remains cooled for the duration of the long-term cooling phase; (2) the boron concentration in the core keeps the core noncritical; and (3) there is no boron precipitation in the core during long-term cooling following a LOCA. Section 15.2.7 of NUREG-1793 describes the staff's evaluation of this long-term cooling analysis. However, the DCD long-term cooling analysis did not consider the effect of debris in the containment on long-term cooling or the GSI-191 evaluation of the AP1000.

As part of the GSI-191 review of the AP1000, the applicant evaluated the effects of containment debris on the long-term core cooling capability. In the AP1000 design, debris could block core flow and adversely affect long-term cooling following a LOCA in three locations: (1) the core inlet; (2) containment recirculation screens; and (3) IRWST screens. Core inlet head loss is the most significant parameter in these analyses because most of the long-term cooling flow and debris goes through the DEDVI and DECL breaks unfiltered into the downcomer and through the core inlet region, and because the AP1000 IRWST and containment recirculation screens are sufficiently large that the debris-induced head loss on the screens is very small. A DEHL break at the reactor vessel adds unfiltered debris into the top of the core region, which does not impact the core inlet loss coefficient. However, it has the potential of flow blockage at the upper part of the core.

The staff's evaluation includes: (1) long-term core cooling analysis sensitivity studies of the effects of the debris-induced head losses at the debris collection locations (i.e., the IRWST and containment recirculation screens) and the reactor core on the long-term cooling analysis and (2) head loss testing of the IRWST and containment recirculation screens and fuel assembly. The applicant's approach is first to perform the long-term cooling sensitivity analysis using the WCOBRA/TRAC code and using arbitrary, nonmechanistic head loss coefficients at the core inlet and the IRWST and containment recirculation screens to determine the maximum allowable head losses in the debris collection locations while still maintaining adequate long-term core cooling. The maximum allowable head losses thus determined serve as the acceptance criteria for the debris-induced head loss tests. The applicant then conducted the

debris-induced head loss testing of the IRWST and containment recirculation screens and fuel assembly to demonstrate that, with the amount of debris collected in the IRWST and containment recirculation screens and the core, the respective test acceptance criteria would not be exceeded. Therefore, one can conclude that adequate core cooling is maintained in the presence of the containment debris. The staff's evaluation of the applicant's submittal appears below.

6.2.1.8.2.7.1 LTC Sensitivity Studies to Determine Adequate Core Cooling

The applicant performed long-term cooling sensitivity studies as documented in TR APP-PXS-GLR-001, Revision 4. As discussed in Section 6.2.1.8.2.1 of this report, Westinghouse selected the DEDVI break for Sensitivity Cases 1 through 10 for the long-term cooling evaluation following a LOCA. All cases were performed using the decay heat assumption in Appendix K to 10 CFR Part 50. As the same long-term cooling case analyzed in DCD Section 15.6.5.4C, Case 1 considers a DEDVI line break in the PXS room with a moderate increase in the core and screen pressure drops resulting from debris blockage. The flow resistance of the lower support plate at the core inlet is increased to model a dP of 3 ft of water at the DCD analysis flow rate of 68.9 kilograms per second (kg/s) (152 pounds per second (lb/s)). Cases 2 and 3 increase these pressure drops. Each of these cases includes added pressure drop across the core, recirculation screens, and IRWST screens. Cases 1, 2, and 3 were performed at the start of containment recirculation at 2.6 hours after a LOCA. By this time, only a small portion of the total debris would have been transported into the RCS, and the debris dP would not have increased to its maximum.

In the sensitivity studies and the containment recirculation sump and IRWST screen testing, the applicant also demonstrated that the screen head losses, considering debris amounts and types, flow rates, and screen size and type, were small, or less than a fraction of an inch of water.

For Sensitivity Cases 4 through 10, the core inlet loss coefficient was further increased to represent different levels of debris plugging at the core inlet. Case 10 had the highest core entrance loss coefficient. The IRWST and containment recirculation screen loss coefficients were assumed to be 0 because they are insignificant compared to the core inlet loss coefficient. Sensitivity Cases 4 through 10 were performed at a later time of 8.6 hours after a LOCA, which is considered the earliest time at which all the debris in the water of the containment can be transported from containment into the core, and increased the dP to its maximum.

In RAI-SRP6.2.2-SRSB-36, the staff asked the applicant to clarify its reason for running Sensitivity Cases 4 through 10 at 8.6 hours after a LOCA. In its response dated January 29, 2010, the applicant stated that for Cases 1 through 3, it assumed that peak core dP would occur earlier, before all particles and fibers would be transported to the core. Sensitivity Cases 1 through 3 were analyzed at 2.6 hours after a LOCA. At this time, only a small amount of debris would start to enter the core through the break. For Cases 4 through 10, the applicant assumed, for purposes of core inlet debris accumulation, that the [ ] would transport all of the particles and fibers. As subsequent fuel assembly head loss tests were run, the applicant noted that the maximum core dP occurred later than the evaluation time of 8.6 hours of debris transport. It therefore concluded that using the decay heat level at 8.6 hours after a LOCA in Sensitivity Cases 4 through 10 was conservative for evaluating fuel assembly test results where maximum core dP is reached 9 hours or later post-LOCA. In its response to RAI-SRP6.2.2-SRSB-40, the applicant demonstrated that subsequent fuel assembly concurrent debris addition test results yielded the peak core dP at much later times after the LOCA than

8.6 hours. Table 3-3 of TR-26, Revision 8, showed that for each of the later fuel assembly debris loading tests, peak core dP at an equivalent plant time far exceeded 8.6 hours after LOCA. The equivalent plant time was determined by comparing the amount of chemical addition before the occurrence of the peak dP in the test with the post-accident chemical effects evaluation. The staff agrees that the sensitivity cases calculated at 8.6 hours are conservative for assuming the maximum core inlet blockage because 8.6 hours is far earlier that the time of the peak debris-induced core inlet dP in the plant. As discussed in the Test Acceptance Criteria section, the core inlet dP for Case 10, which was a DEDVI line break in the loop compartment with the largest core inlet resistance among all sensitivity cases, will be used as the acceptance criterion for the fuel assembly debris blockage head loss tests after 9 hours. The core inlet dP for Case 3 will be used as an additional acceptance criterion for the fuel assembly head loss test before 9 hours.

For Sensitivity Case 11, the applicant modeled a DEDVI break in the loop compartment close to the reactor vessel DVI nozzle, simulating significant resistances at the core exit region. The added debris resistance was applied to the core exit in order to provide insights into the impact of containment debris entering the upper part of the core during a postulated HL break LOCA. Because postulated debris would be introduced into the upper plenum during a HL break scenario, no increase in the core entrance flow resistance above the value associated with normal plant power operation was modeled. This case was executed at the decay heat level at 8.6 hours post-LOCA time to transport sump debris into the reactor vessel. The results of Case 11 provide acceptable values of core flow and debris-induced dP for application to fuel assembly head loss testing that evaluated DEHL LOCAs.

The applicant used WCOBRA/TRAC and WGOTHIC for the RCS transient and containment analyses, respectively, of the long-term cooling analysis, as documented in DCD Tier 2, Section 15.6.5.4C, and for the 11 sensitivity cases evaluated in APP-GW-GLR-001. The long-term cooling analysis used a detailed nodalization model to represent the AP1000 core for the WCOBRA/TRAC analysis. WCAP-14776, "WCOBRA/TRAC, OSU Long-Term Cooling Final Validation Report," Revision 4, issued March 1998, documents the code verification for the long-term cooling analyses. The applicant used the WGOTHIC code, described in WCAP-15846, Volume 1, Revision 1, to calculate containment boundary conditions. The fan coolers were assumed to be operating to minimize containment pressure. The staff previously approved the application of WCOBRA/TRAC and WGOTHIC for long-term cooling calculations performed in the AP1000 DCD, as discussed in Chapter 21 of NUREG-1793.

The staff reviewed the application of WCOBRA/TRAC and WGOTHIC for performing the sensitivity studies documented in APP-GW-GLR-001. The staff conducted an audit in Monroeville, Pennsylvania, September 21–22, 2009, and a followup audit March 22–24, 2010. During the September 2009 audit, the staff identified discrepancies between the WCORBRA/TRAC and WGOTHIC calculations for sump water temperature input. In RAI-SRP6.2.2-SRSB-25, the staff asked the applicant to address these inconsistencies.

In its response, the applicant stated that the WGOTHIC containment analysis case was reanalyzed using corrected revised input, and the calculated sump water temperatures were within -17.22 °C (1 °F) of the corresponding values presented in the WCOBRA/TRAC calculation. The applicant stated that since the boundary conditions for WCOBRA/TRAC from the new WGOTHIC case differed minimally, the conclusions of Sensitivity Cases 1 through 3 in APP-PXS-GLR-001 were not affected.

During the audit performed March 22–24, 2010, the staff reviewed the new WGOTHIC analyses and results used for Sensitivity Cases 4 through 10. The applicant also responded to RAI-SRP6.2.2-SRSB-25, Revision 1, which addressed the WGOTHIC containment pressure and sump temperatures used in the Sensitivity Case 1 through 10 long-term cooling analyses. The pressure used in the long-term cooling calculation was 110 kPa (16 pounds per square inch absolute (psia)), which is lower than the pressure calculated by WGOTHIC. The sump temperatures calculated by WGOTHIC were slightly different from the sump temperatures used in the DCD Revision 17 long-term cooling calculations. The applicant has established that a lower pressure is conservative for long-term cooling calculations and that the small changes to the sump temperature do not impact the long-term cooling analysis. Based on its review of the new WGOTHIC analysis and the information provided in response to RAI-SRP6.2.2-SRSB-25, Revision 1, the staff concluded that the analyses were performed in a conservative manner and were acceptable.

The more limiting long-term cooling sensitivity cases in APP-GW-GLR-001 assume large core inlet head losses to simulate extreme core inlet blockage by the containment debris. These cases result in low core flow and high steam quality, which differ from the DCD long-term cooling analysis. The staff, in RAI-SRP6.2.2-SRSB-39, asked the applicant to confirm that the WCOBRA/TRAC code had been validated for the low flow and high steam quality conditions.

In its response dated January 29, 2010 (ADAMS Accession Number ML100330392), the applicant indicated that the validation of WCOBRA/TRAC is provided through the comparison of WCOBRA/TRAC simulations to test data from boiloff tests documented in WCAP-15644-P, "AP1000 Code Applicability Report," Revision 2, issued March 2004. During the audit performed March 22–24, 2010, the staff reviewed the range of applicability of boiloff tests G1 (WCAP-9764, "Documentation of the Westinghouse Core Uncovery Tests and Small Break Evaluation Model Core Mixture Level Model," issued July 1980) and G2 (Andreychek, T.S., "Heat Transfer above the Two-Phase Mixture Level under Core Uncovery Conditions in a 336 Rod Bundle," Volumes 1 and 2, Electric Power Research Institute (EPRI) Report NP-1692, issued January 1981) to the conditions that would be simulated for Sensitivity Cases 4 through 10. In WCAP-15644-P, the applicant used WCOBRA/TRAC to simulate selected G1 and G2 boiloff tests, at low pressure, low flow and a range of power levels typical of AP1000 LTC conditions. WCOBRA/TRAC tended to overpredict the level swell in the core during boiloff scenarios. However, when a multiplier (YDRAD=0.8) is applied to the interfacial drag coefficient computed from the vertical flow regime models in the code, WCOBRA/TRAC generally predicted the level swell to within +/- 20 percent of the measured value. Therefore, the YDRAG=0.8 was selected for use in the WCOBRA/TRAC model for the AP1000 LTC analysis. The staff reviewed the ranges of the G1 and G2 test conditions and found that the conditions simulated for the G1 and G2 tests by WCOBRA/TRAC bounded the conditions calculated in Sensitivity Cases 4 through 10.

The staff noted in its review that the validation evaluations were performed using the WCORBRA/TRAC M7AR4_SB03 code version, whereas the current long-term cooling sensitivity analysis cases were performed using WCORBRA/TRAC M7AR7_AP. The staff asked if the difference in the different code versions could invalidate the conclusions of the previous studies to validate that WCORBRA/TRAC can successfully calculate conditions seen in the high debris blockage conditions of Sensitivity Cases 4 through 10. In Revision 2 of its response to RAI-SRP6.2.2-SRSB-39, dated June 28, 2010, the applicant indicated that it reviewed the differences between the two code versions and found that all but one of the error corrections and code updates can be classified into one of the four categories judged to have no or negligible impact on the applicability of the G1 and G2 validation calculations to the long-term

cooling debris sensitivity cases. The only code difference judged to potentially affect the level swell calculations is related to the use of a level sharpener in the WCORBRA/TRAC M7AR4_SB03 code version. The WCOBRA/TRAC_AP code does not contain an explicit mixture level tracking model and, therefore, tracking of two-phase mixture level is accomplished by nodalization and prediction of the axial void gradient between hydraulic cells. The "level sharper" model in WCOBRA/TRAC M7AR4_SB3 code version was developed to locate the mixture level in the hydraulic cells where a sharp void fraction gradient is detected in the vicinity of the two-phase mixture level. The level sharpener logic is only applied to the void fraction used in the fuel rod heat transfer calculations in the hydraulic cells where detailed representation of local void fraction is important to assure that fuel rod heat transfer is computed based on the appropriate fluid condition. The level sharpener does not directly affect the global void fraction distribution. Since the level swell is a measure of two-phase mixture level relative to the collapsed liquid level, and is calculated in terms of the average void fraction, the precise mixture level has minimal effect on the level swell calculation. Therefore, the level sharpener model has negligible effect on the level swell calculation, and its impact on the G1 and G2 simulations is small. In other word, similar conclusions for the WCOBRA/TRAC code validation with the G1 and G2 simulations can be drawn with and without the level sharpener model. Therefore, the staff concludes that the use of the interfacial drag coefficient multiplier identified with the code version with the level sharpener model is applicable to the code version without the level sharpener logic. It should be noted that the WCOBRA/TRAC-M7AR4_AP code version used in the DCD Revision 15 long-term cooling analysis also did not include the level sharpener logic.

Sensitivity Cases 1 through 3 were analyzed with the AP1000 reference core design described in DCD Chapter 4. During the March 22–24, 2010, audit, the staff found that Sensitivity Cases 4 through 11, documented in Westinghouse calculation note APP-SSAR-GSC-732, "AP1000 AFCAP Post-LOCA Long-term Core Cooling Analysis," Revision 0, dated November 12, 2009, were performed with an advanced first core design, which differs from the DCD reference core design. Since the Advanced First Core Analysis Program (AFCAP) core design differs from the core design described in the AP1000 DCD, the staff questioned the applicability of the LTC analysis for cases 4 through 10 to the DCD core design. In response to RAI-SRP6.2.2-SRSB-42, the applicant describes the differences between the AFCAP core design and the DCD core design. The majority of the fuel assembly characteristics are either unchanged or have trivial changes in the AFCAP core design, [

]. These changes are reflected in the input of the WCOBRA/TRAC AFCAP analysis for LTC sensitivity study Cases 4 through 10 using larger values of the flow resistance form loss coefficients for the mixing vane grids (MVG) and intermediate flow mixer (IFM) grids in the active fuel region. Other parameters in the WCOBRA/TRAC LTC analysis remain unchanged between the AFCAP and the DCD reference core design. The WCOBRA/TRAC LTC analysis was performed with the core geometry and form loss coefficients of fuel assemblies, plus a large non-mechanistic form loss coefficient at the core inlet to simulate possible worst case debris plugging. Since the core inlet flow resistance used in the LTC sensitivity runs are significantly larger than the flow resistances of the MVG and IFM, the differences in the MVG and IFM flow resistance between the AFCAP and DCD core design is not significant. In addition, the use of higher resistances for the MVG and IFM for the AFCAP core design would result in lower core flow rate than the DCD core design with lower MVG and IFM resistances. Since the objective of the LTC sensitivity studies is to establish the flow/pressure drop acceptance criteria for the AP1000 debris-induced core inlet blockage tests, the analysis using the AFCAP core design would result in lower core flow rate and inlet pressure drop. This results in more restrictive acceptance criteria for the fuel assembly debris-induced head loss tests. The use of the more restrictive acceptance criteria

from the AFCAP core design for the fuel assembly head loss tests of the DCD referenced core design is conservative and is, therefore, acceptable.

LTC Sensitivity Study Results

Section 3 of APP-GLR-PXS-001, Revision 4, presents the results of the long-term cooling sensitivity analysis. For the DEDVI line break, the effects of increasing the flow resistance at the core entrance are generally reflected in the increase in the downcomer liquid level, a decrease in the core flow rate, a decrease in the core collapsed liquid level, an increase in the upper core void fraction, and an increase in the quality of the discharge flow through the ADS-4 valves. The upper plenum pressure for each case essentially reflects the containment pressure. In all cases analyzed, the collapsed liquid level in the core is higher than or near, the core midplane. The mean HL collapsed liquid level is above the HL centerline. For each sensitivity case, Table 4-1 in APP-GLR-PXS-001, Revision 4, provides the time after LOCA, core inlet loss coefficients, core flow rate, core inlet dP, ADS-4 steam quality, and maximum core boron concentration.

For Sensitivity Case 3, the upper plenum pressure of 22 psia is essentially unchanged from the DCD case. The downcomer liquid level has increased compared to the DCD case because of added core inlet flow resistance. Injection rates through the DVI lines into the vessel are reduced compared to the DCD case values. Flow through the intact DVI line is reduced to 25 kg/s (56 lbm/s) versus 35 kg/s (77.2 lbm/s) in the DCD analysis. Flow into the vessel through the broken DVI line is reduced from 34 kg/s (75 lbm/s) in the DCD analysis to 25 kg/s (55 lbm/s) in Sensitivity Case 3. The core flow is reduced to 50 kg/s (111 lbm/s) with a pressure loss of 24.1 kPa (3.5 pounds per square inch (psi)). The HL collapsed liquid level is above the HL centerline. The quality of the flow discharged through ADS-4 valves in Sensitivity Case 3 is approximately 0.35. The core boron concentration is 4,700 parts per million (ppm).

The thermal hydraulic behavior in the core was shown to be most limiting in Sensitivity Case 10. For Sensitivity Case 10, the downcomer liquid is established at a higher level than in the DCD analysis because of the greater added resistance at the core entrance. Boiling in the core produces steam and a two-phase mixture that flows out of the core into the upper plenum. The core collapsed liquid level is maintained at a mean level above or near the core midplane even with the added core entrance resistance. Boiling causes pressure variations, which in turn cause variations in the core collapsed level and the flow rates of liquid and vapor from the top of the core. The HL collapsed liquid level is around the HL centerline. The peak cladding temperature of the hot rod closely follows the saturation temperature. The flow through the core and out of the RCS provides adequate flushing to preclude the unacceptable concentration of the boric acid solution. Liquid collects above the upper core plate in the upper plenum, where the collapsed liquid level remains well above the active fuel length.

For Sensitivity Case 10, the upper plenum pressure reflects the prevailing containment pressure of 110 kPa (16 psia) calculated at 31,000 seconds. Injection rates through the DVI lines into the vessel are reduced compared to the DCD case values, and the injection flow rate is greater through the lower resistance broken DVI line than it is through the intact DVI line. The core flow is predicted to be 29.5 kg/s (65 lb /s), with a pressure loss of 28.3 kPa (4.1 psi). The steam quality of flow discharged through ADS-4 valves is approximately 0.49. This liquid carryover is adequate to limit the concentration of boric acid in the core water to a value of 6,100 ppm. This is lower than the maximum value of 7,400 ppm at the time of recirculation initiation in the DCD long-term cooling analysis that was shown to be acceptable in DCD Section 15.6.5.4C.4.

The applicant determined that Sensitivity Case 10, which is a DEDVI break in the loop compartment, represented the worst case condition assuming that all the debris will reach the core entrance sometime after one complete recirculation of the entire containment water volume.

The applicant used Cases 3 and 10 to establish the acceptance criteria for the AP1000 fuel assembly debris head loss testing. The staff's evaluation of these acceptance criteria is discussed in the Test Acceptance Criteria section of this SER section.

For Case 11, the resulting injection flow rate through the lower resistance broken DVI line 62.8 kg/s (138.5 lb /s) is greater than it is through the intact DVI line 34.5 kg/s (76 lbm/s). The core flow is predicted to be 97.3 kg/s (214.5 lb /s) with an average pressure loss of 13.8 kPa (2 psid) at the core exit where the added debris resistance was applied. The results of Case 11 also demonstrate long-term cooling performance comparable to the DCD long-term cooling case. The DEHL break acceptance criterion is core exit pressure drop of 13.8 kPa (2 psid) at a flow of 97.3 kg/s (214.5 lbm/s).

It should be noted that the WCOBRA/TRAC long-term cooling analysis did not consider the effects of the potential accumulation of noncondensable gases in high points in the PXS flowpaths. In a letter dated May 25, 2010, the applicant proposed the PXS design changes (Change No. 66), including installation of high point vents, to reduce the potential impact on gas intrusion in the PXS flowpaths. The staff considers that an appropriate PXS design change to prevent potential gas accumulation in the PXS system is an acceptable substitute for not considering the noncondensable gas effects in the long-term cooling analysis. The staff evaluation of DCP Change No. 66 will be addressed in Chapter 23.

6.2.1.8.2.7.2 Fuel Assembly Head loss Testing

The applicant performed a series of experiments to quantify the effect of fibrous and particulate debris and containment chemical effects on the head loss across the fuel assemblies of an AP1000 core during a postulated LOCA, documented in WCAP-17028, Revision 6, and performed in consideration of GSI-191. The objective of these experiments was to demonstrate that there is reasonable assurance that the AP1000 can provide adequate post-LOCA long-term core cooling. The applicant used a fuel assembly design that is consistent with the fuel assembly design described in the AP1000 DCD. The flow rates, debris loadings, and method of debris addition varied from test to test. The ratio of fibrous to particulate debris varied, as did the temperature and chemistry of the coolant. The purpose of the tests was to select a combination of debris variables and simulated plant variables that would bound any AP1000 LOCA and to demonstrate available long-term cooling margin in the AP1000 design such that with the flow blockage caused by the containment debris transported to the reactor vessel, the head losses determined by the fuel assembly testing are bounded by the test acceptance criteria described below.

The applicant performed 39 different experiments. Section 7 and Tables 7-1 through 7-3 of WCAP-17028 summarize the test matrix and initial conditions for the tests. Table 8-1 of WCAP-17028 summarizes the results of experiments that were performed with various debris loads, flow rates, fiber length, chemical effects, and fiber, particulate, and chemical addition sequencing.

The applicant had initially conducted the first 16 fuel assembly tests correlating to long-term cooling Sensitivity Cases 1 through 3. The applicant varied the particulate, fiber, and chemical

amounts. The composition of the fibers for these 16 fuel assembly tests also varied. For the subsequent 23 fuel assembly tests, except for tests 35 and 38, the applicant increased fibrous debris loading based on the design-basis fiber amount of 3.0 kg (6.6 lb) in the containment, equivalent to 2.7 kg (6 lb) of fiber to the core inlet, as discussed in Section 6.2.1.8.2.6 of this report. The applicant chose debris loadings in the fuel assembly tests to bound the quantities that could be transported to one fuel assembly in the AP1000. Table 3-4 of TR-26 presents the AP1000 design-basis fibrous, particulate, and chemical debris that could be transported into the reactor vessel and possibly reach the fuel assemblies.

The applicant selected flow rates to bound the conditions expected during post-LOCA recirculation core cooling. Except for Tests 2 and 8 through 11, the first 16 tests were conducted with constant flow rates. Tests 17 through 39 were conducted with variable flow rates, with the flow reduction based on the increase in the dP resulting from debris additions.

Tests 1 through 21 and Test 23 were performed using sequential additions of particles, then fibers, then chemical surrogate. Test 22 and Tests 24 through 39 were performed with concurrent additions of particles, fibers, and chemical surrogate. The applicant prepared fibrous and particulate debris loads and chemical precipitates outside of the test loop and added them to the makeup tank in accordance with the applicant's test plan. All of the experiments included fibrous and particulate debris and chemical reaction products that were added to the makeup tank as water suspensions. The sequential and concurrent debris addition schemes are discussed in the Debris Addition Scheme section below.

The fuel assembly test loop consists of a scaled length of a fuel assembly inside a Plexiglas case, a mixing tank, and the pumps and plumbing required for circulating water and debris. The fuel assembly is consistent with the design described in DCD Section 4.2.2.2. In RAI-SRP6.2.2-SRSB-26, the staff asked the applicant to justify why the test results from the scaled-length, isothermal fuel assembly test facility are applicable to the post-LOCA long-term cooling situation for a full-length assembly in which boiling occurs in the upper portion of the fuel assembly. In its February 26, 2010, response, the applicant stated that most of the AP1000 fuel assembly debris head loss tests were performed with a single fuel assembly and upflow of water. The applicant included a fuel bottom nozzle, p-grid, and spacer grids in the single fuel assembly, simulating the bottom portion of a fuel assembly. The results from the fuel assembly tests have shown that the debris-induced pressure drop (dP) acceptance criteria are met. The upflow fuel assembly tests simulated DEDVI or DECL LOCAs where the debris entered into the downcomer and the core inlet region.

The fuel assembly test results show that the vast majority of the dP was seen across the inlet nozzle and p-grid in the bottom part of the test assembly; therefore, the applicant concluded that the use of a scaled-length test assembly will not change the test results because most of the pressure drop occurs in the first part of the test assembly. On the basis of the test results that show the majority of the dP occurs in the inlet nozzle and p-grid in the bottom part of the test assembly, the staff agrees that the scaled-length test fuel assembly is reasonable to simulate the AP1000 fuel assembly for debris blockage at the core inlet.

The applicant set up a test loop, as described in Appendix C to WCAP-17028, and configured it in two ways: (1) to simulate a DVI break or CL break with break flow through the downcomer and up through the core inlet region; and (2) to simulate a HL break with downflow from the break to the core exit region. The test facility also simulated additional conditions, such as water at higher temperatures and the boiling environment seen in the AP1000 core when there is a high level of debris blockage at the core inlet. The boiling environment in the core is

simulated by injection of air during the test. The test loop flow rates were scaled to expected flows seen in the AP1000 to represent the flow rate at the core inlet for a DECL/DVI break with upflow. The selected flow rates were representative of or bounded flows expected through the core during the recirculation phase after a LOCA.

In response to RAI-SRP6.2.2-SRSB-26, the applicant performed additional fuel assembly Tests 36, 37, and 39 to address the applicability of the isothermal scaled test to potential boiling conditions at the full-length fuel assembly. The staff asked whether there are situations where two-phase flow could challenge the single-phase test results and whether a higher liquid temperature or local boiling phenomenon could affect the behavior of the debris plugging the core. Tests 36 and 37 repeated Test 30, with the exception that chemicals were added to the test loop to simulate reactor coolant chemistry and the test loop was heated to a higher temperature than for Test 30. The applicant's test results showed, as before, that the fiber accumulated around the bottom of the p-grid before accumulating around the bottom nozzle. For both tests, near the end of the flow sweeps, the debris bed was very thick and very smooth, yet it did not cause an increase in the steady-state core dP. The pressure drop increase occurred almost exclusively at the bottom nozzle/p-grid location as in the other tests. On the basis of its review, the staff agreed that using the lower temperature water in the test facility in Test 30 was a conservative effect on the measured core inlet dP. As a result, the staff found that testing with lower temperature water was acceptable.

The applicant performed an additional Test 39 in response to RAI-SRP6.2.2-SRSB-26 to observe the effect of adding debris to the test fuel assembly under simulated DEHL break conditions. The conditions simulated in this test were upward flow of coolant with boiling. The debris loadings used were the same as for Test 30. [

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In the WCOBRA/TRAC analysis, the applicant predicted significant two-phase mixture at the top of the AP1000 core. In APP-GLR-PXS-001, Sensitivity Case 10 was determined to be the worst case debris blockage case simulated by WCOBRA/TRAC. The applicant found that the pressure drop across the core for single-phase flow at the core inlet was larger than the pressure drop across the top part of the core even when two-phase flow at the top of the core causes the flow losses to be much greater.

On the basis of its review, the staff finds that the fuel assembly debris loading head loss testing using single-phase flow appropriately estimates the worst case dP across the core inlet when compared to the two-phase flow predicted in an AP1000 core after a LOCA.

Implicit in the single fuel assembly debris load head loss test is a basic assumption of uniform blockage across the core inlet. In RAI-SRP6.2.2-SRSB-33, the staff requested that the applicant discuss potential effects of non-uniform blockage in the fuel assembly test facility. In its response, dated January 29, 2010, the applicant explained that the AP1000 fuel assembly debris-loading head loss tests indicate that there is non-uniform flow blockage and there is considerable variation in the fuel assembly flow/dP even with the same debris addition. Every test has shown non-uniform blockages within the fuel assembly and gaps in the debris bed. This shows that the debris bed at one fuel assembly will be different from that in others.

The applicant further stated that it is expected that the distribution of debris across the core inlet of all fuel assemblies will be non-uniform. Debris that accumulates at the core inlet is not expected to distribute uniformly across the entire core inlet and, therefore, some fuel assemblies will have more debris buildup than other fuel assemblies. The fuel assemblies with less debris buildup will experience less dP across the fuel assembly and will be able to pass more flow through those fuel assemblies. The higher flows through these low dP fuel assemblies would cross over to assist cooling fuel assemblies that have higher dP and lower flow at the core inlet and provide additional margin for long-term cooling. Therefore, the staff concludes that the single fuel assembly debris-induced head loss test is conservative and acceptable.

Debris Addition Scheme

The manner in which the debris is added has been observed in prior industry testing programs to make a difference in the overall head loss through the fuel assembly. Therefore, it was determined that the order in which particulate, fiber, and chemicals were added was important. the applicant used the following debris addition schemes (i.e., sequential addition and concurrent addition) for the fuel assembly debris loading head loss tests:

- For tests applying the sequential additions of particulate, fiber, and chemicals, an aliquot of the water from the mixing tank was removed and placed in a container for each addition of particulate. The particulate was then well mixed into this water until completely suspended before being added to the mixing tank. The mixing tank volume was allowed to thoroughly mix. Then the fiber was added per the test plan using a similar manner as described above for the particulate. Thirdly, the surrogates for chemical reaction products were added to the test loop after all of the fiber and particulate had been added. The chemical precipitate was mixed outside the test loop per the WCAP-16530-NP-A methodology and then added to the test loop in measured batches. The approach of sequential debris addition into the test loop is consistent with the NRC-approved guidance on head loss testing, WCAP-16406-P-A, Revision 1. The fuel assembly tests using sequential debris addition were set up to determine the maximum dP value that would be achieved under the test condition.

- Tests applying the concurrent addition of particulate, fiber, and chemicals modeled debris addition at start of recirculation after a LOCA. For tests applying concurrent debris addition, an aliquot of the water from the mixing tank was removed and placed in a container and the prescribed amounts of particulate and fiber were added to the container. The particulate and fiber were then well mixed in this water until completely suspended before being added to the mixing tank. Concurrent with the addition of particulate and fiber, chemical surrogate was added directly to the mixing tank in the amount prescribed in the test procedure. Concurrent debris additions were made at times specified in the test procedure.

Test Acceptance Criteria

In the long-term cooling sensitivity studies described above, Case 10 provided the limiting case conditions, which were postulated assuming that all the debris will reach the core entrance after one complete recirculation of the entire containment water volume. For Case 10, the long-term core cooling analysis found that for a CL or DVI break, the limiting dP through the core is 28.3 kPa (4.1 psid) with a corresponding minimum core flow rate of 29.5 kg/s (65.0 lb /s or 480.7 gallons per minute (gpm) core flow rate), or [ ] per fuel assembly. Therefore, the acceptance criterion for the fuel assembly test is 28.3 kPa (4.1 psid) at [ ] per fuel

assembly. This is identified as acceptance criterion 1 for all fuel assembly tests not involving a HL break.

The second criterion is based on sensitivity Case 3 in the long-term cooling analysis. In this case, the maximum dP allowed in the core at the beginning of the recirculation phase is 24.1 kPa (3.5 psi) with a flow rate of [ ] per fuel assembly. The second criterion is intended to verify that during the recirculation, for any level of the decay heat between the start of the recirculation phase (long-term cooling Sensitivity Case 3) and 8.6 hours, the core will be satisfactorily cooled, requiring that the head loss through the core at [ ] per fuel assembly must be lower than 24.1 kPa (3.5 psi). It should be noted that the concurrent debris addition fuel assembly tests were set up in order to model the plant timing of debris addition after the start of recirculation. Therefore, the test time can be related to the plant time. In contrast, the sequential debris addition tests were set up to determine the maximum dP value that would be achieved under the test condition, with no reference to the time at which it would occur in the plant. For this reason, the second acceptance criterion is not applicable to the sequential debris addition tests but only to the concurrent debris addition tests.

For those fuel assembly debris head loss tests where the debris is added in a mechanistic, time-dependent sequence for the beginning of recirculation (2.6 hours) after a DECL or DEDVI LOCA, the applicant used Sensitivity Case 3 for the core dP acceptance criteria. This case is conservative because: (1) debris-induced resistance at the core inlet is assumed to occur at the beginning of recirculation, earlier than the time debris would be transported to the core inlet; and (2) the decay heat rate at 2.6 hours post-LOCA is higher than the decay heat rate at 9.0 hours, the time at which the Sensitivity Case 10 acceptance criteria are used.

For a HL break the flow in the upper part of the core is expected to oscillate when the downflow in a low-power assembly becomes low and the steam generation rises upwards. Case 11 from the sensitivity studies found that the limiting dP through the core is 13.8 kPa (2 psid) with a corresponding minimum flow rate of 97.3 kg/s (214.5 lbm/s or 1,602 gpm core flow rate). This is the minimum flow rate that can be maintained for long-term core cooling. The acceptance criterion for the HL break fuel assembly tests is that the dP across the fuel assembly test article must not exceed 13.8 kPa (2 psid) at a flow rate of [ ]. This acceptance criterion will be applied to each of the HL break fuel assembly tests.

As discussed earlier, there are two acceptance criteria for the upflow fuel assembly tests simulating CL and DVI line breaks. The first acceptance criterion for the fuel assembly tests is that dP across the fuel assembly inlet must not exceed 4.1 psid at a flow rate of [ ], which is based on the limiting Case 10 of the long-term cooling sensitivity studies. This acceptance criterion was applied to each of the fuel assembly tests. The second criterion, applied to all concurrent debris addition tests, is intended to verify that during containment recirculation when the debris bed is not completely formed, for any decay heat rate between the start of the recirculation phase at 2.6 hours (long-term cooling Sensitivity Case 3) and the time of 8.6 hours that it takes to recirculate one containment water volume, the core will be adequately cooled, with the core head loss lower than 24.1 kPa (3.5 psid) at fuel assembly flow rates of [ ]. Another test acceptance criterion for a HL break is 13.8 kPa (2 psid) at a flow rate of [ ] through the fuel assembly.

The applicant conducted many CL tests with a constant flow rate higher than [ ] or with variable flow rates where the minimum flows were higher than [ ]. Moreover, in all the concurrent addition tests, the flow rate at a time before 9 hours in plant time was higher than [ ]. To evaluate the test results, the experimental results need to be adjusted to a lower

flow rate ([ ] for the first criterion, [ ] for the second one). The applicant developed Equations 5.1.2 and 5.1.3 in WCAP-17028 to calculate the adjusted dP to flow rates of [ ] and [ ], respectively, for the comparison with the first and second acceptance criteria. These equations were developed based on the dP/flow rate relationship developed for each fuel assembly test described in Section 5.2 of WCAP-17028. The exponent "b" in Equations 5.1.2 and 5.1.3 is determined from the fuel assembly head loss tests and summarized in Table 5-1 of WCAP-17028 for each test. In Section 8.39 of WCAP-17028, the applicant described the development of Equations 5.1.2 and 5.1.3, as well as the value of exponent b in these equations.

In RAI-SRP6.2.2-SRSB-27 and RAI-SRP6.2.2-SRSB-29, the staff asked for information about the development of the b exponent in extrapolating the expected pressure differential and the use of the two acceptance criteria for the testing program. In its February 26, 2010 response, the applicant explained that many of the AP1000 tests were conducted with variable flow rates where the flow was changed during the test as the dP increased to simulate the actual behavior of the plant. Therefore, the applicant developed two acceptance criteria for the fuel assembly debris load tests. This was discussed in WCAP-17028 in Sections 5 and 8.

As discussed above, to determine whether a test met the long-term cooling criteria, the maximum dP measured in each test was adjusted, as described below, based on both the flow that existed when the maximum dP occurred and the minimum acceptance flow (i.e., [ ]). This adjusted dP provides a simple way to compare tests and determine whether the test met the acceptance criteria.

The data collected by the tests resulted in the development of Equation 8.39.1 in WCAP-17028, which defines the relationship between head loss and the flow rate. This equation is based on the Darcy formula, and the exponent is determined by test results. When the debris bed is formed and stable, the pressure drop behavior of the debris bed will vary consistently with flow rate. In other words, [ ]. This also means that once the value of the [ ] are known at a particular flow rate, it is possible to evaluate the value of dP at any flow rate.

Since the tests were not conducted at a low flow rate of [ ] as required by the first acceptance criterion, the measured dP must be adjusted as suggested above. For the first acceptance criterion, some of the tests performed flow sweeps to the low flow level. From each of these tests, an exponent was developed and used in Equation 8.39.4 of WCAP-17028 to calculate the pressure differential adjusted to the low flow of [ ]. To illustrate this process, Test 34 will be reviewed below.

Test 34 was performed to investigate the nature of the dP/flow relationship throughout the test to allow comparison of the bed behavior for a fully formed and stable debris bed. Flow sweeps were performed throughout the duration of Test 34, and the experimental results confirm that the dP and the flow are related by a power law relationship as shown in Equation 8.39.1 of WCAP-17028 even in the case of a debris bed not yet fully formed. [ ], which was much lower than the first acceptance criterion of 28.3 kPa (4.1 psid). The test dP at the maximum debris bed resistance as presented in Table 8-1 was used for the first acceptance criterion. Since Test 34 was a concurrent test, the second acceptance criterion must also be applied.

The second criterion is based on long-term cooling Sensitivity Case 3, which assumes the maximum blockage condition in the core at the beginning of the recirculation.

Concerning the applicability of Equation 8.39.3 to the second acceptance criterion, the use of the stable bed exponent would result in a greater underestimation of the adjusted dP. Looking at the results of Test 34, the exponent at [ ], and the bed resistance is still increasing. This suggests that a reduction of about 27 percent should be applied at the stable bed exponent to extrapolate the test results at higher flow rates down to [ ], which is the flow basis for the second criterion. For conservatism, a reduction of [ ] was applied to the exponent b of the second criterion. This reduction is based on the difference between the fully formed bed exponent and the lowest exponent estimated in Test 34. For the second acceptance criterion, the exponent b becomes [ ] because of the reduction applied to the exponent. Applying the second criterion to Test 34, the b exponent for Test 34 was determined to be [ ]; applying Equation 8.39.5 with a measured maximum dP at [ ], which was much lower than the second acceptance criterion of 24.1 kPa (3.5 psid).

For most tests, the value of the exponent applied to flow was determined by the flow/dP data taken from that test. For Tests 18 through 34 and Tests 36 and 37, flow sweeps were conducted at the end of the test that provided many data points (different flows/dPs). The flow sweeps reduced the flow rate to the acceptance criteria flow so that these data are directly applicable. Such data are also available for Tests 8 through 11 because of the use of oscillating flows during the tests. Tests 1 through 6 and 13 through 16 used the average exponent from all tests. Table 9-1 and Figure 9-1 of WCAP-17028 show these results. The data show considerable margin between the scaled dP and the acceptance criterion. For example, for Test Case 33, which has the highest scaled dP, there is close to 50 percent margin from the acceptance criteria.

In the AP1000, debris can transport into the RCS though the flooded HL and possibly into the upper parts of the core. In an actual HL LOCA in an AP1000, the flow in the upper part of the core is expected to oscillate over several minutes. When the downflow in a low-power assembly becomes low, the steam generation rises. The tests before Test 35 were all conducted simulating a CL break scenario and without simulating boiling. In RAI-SRP6.2.2-SRSB-31, the staff asked the applicant for more information and clarification concerning the HL break. In response to this RAI, the applicant performed more testing to simulate the effects of a HL break on the AP1000 core. These tests were conducted to prove that the CL breaks were more limiting even if debris entering the top of the core would occur in a HL break. These conditions were to be tested for an expected HL break LOCA in an AP1000.

In the event of a HL break, the applicant assumed that the post-LOCA containment debris load would be equal to 90.72 kg (200 lbs), consisting of 3.0 kg (6.6 lbs) of fiber, 87.72 kg (193.4 lbs) of particulates, and 25.85 kg (57 lbs) of chemical precipitates. The applicant assumed that all the debris was transported into the core through the outer ring of fuel assemblies.

Tests 35, 38, and 39 represented the HL breaks. The goal of Test 35 was to observe the effects of adding debris to a model fuel assembly under reverse flow conditions that might exist in an AP1000 reactor after a HL break. The test evaluated the distribution of debris blockages within the top portion of the assembly, and the relationship between pressure and flow. To model AP1000 behavior on a HL break, further testing of debris blockage with reverse flow was needed. With the debris load and chemical precipitate tested, the peak head loss recorded for

this test was [ ]. The [ ] psid was below the acceptance criteria of [ ].

The purpose of Test 38 was to investigate the debris behavior in the outer fuel assemblies under simulated DEHL LOCA conditions, where flow could be downward and transport debris into the upper part of these fuel assemblies. The test also investigated the impact of changing the direction of flow from the downward to the upward direction, representing steam with boiling present in the upward direction. Test 38 was conducted to represent the downflow in a low-power assembly that produces steam and results in a change in flow from the downward to upward direction. This flow reversal and its effect on a debris bed were explored. The downflow was introduced for approximately 1 hour after one concurrent debris addition. The flow was then reversed to upflow, simulating the steam generation rise, and boiling was introduced at this time.

The reversal of flow did not visibly change the bed a great deal, but when air was introduced to simulate boiling, the [ ]. The air was injected at [ ], which is the maximum allowable air injection for the test rig and is much lower than the steam volume predicted to exist in the upper core region, and proved that this flow rate can quickly disperse any debris that would collect. Therefore any flow rate greater than [ ] would do the same, but more quickly. Based on the test results, the applicant concluded that the brief upflow of steam would be sufficient to break up any accumulated debris in the upper core region. The purpose of Test 39 was to investigate the debris behavior in central fuel assemblies that will be exposed to constant upward flow of water and steam. Test 39 was conducted solely to represent the HL break condition representing the steam in the upward flow direction and the local boiling phenomenon affecting the behavior of the debris plugging the core. The air was injected at the same rate in Rest 38 throughout the duration of the test. [ ].

Tests 38 and 39 had very low pressure drops, [ ].

[ ]. The pressure drops in the core for all three HL tests were less than the acceptance criterion of 13.8 kPa (2 psid).

In RAI-SRP6.2.2-SRSB-28 and RAI-SRP6.2.2-SRSB-30, the staff asked the applicant to explain the large variation in test results for Tests 27, 29, and 30, given they had the same amount of debris and debris addition procedures and appeared to be repeatable tests. The staff also noted that the fuel assembly test results indicate large uncertainties where the peak dP is significantly different for similar flow cases with the same amount of fiber. The staff asked the applicant to justify the acceptability of test results with large uncertainties and to provide an evaluation of the statistical confidence with which the test results could be used to assess the long-term cooling effectiveness based on the fuel assembly debris loading head loss tests.

In its response, the applicant prepared APP-GW-GLR-092, Revision 0, describing its statistical analysis of the fuel assembly debris loading head loss tests. The objective of the statistical analysis was to use available test data to show that there is a low probability that the AP1000 debris bed resistance will exceed the analyzed safety analysis limit from the long-term cooling sensitivity studies, which show acceptable results for the DVI break or CL break scenarios where the debris enters the core from the downcomer and lower plenum. The applicant concluded that, based on the test data and consistent with the statistical evaluation, it had established a conservative distribution of the adjusted pressure drop across the core. Using this conservative distribution, the effective adjusted pressure drop at the core inlet was calculated to be significantly below the safety analysis limit of 28.3 kPa (4.1 psid) with core flow of 29.5 kg/s (65 lbm/s).

The statistical analysis of the tests evaluated in APP-GW-GLR-092, Revision 0, determined that the probability for a single fuel assembly to exceed the acceptance criterion of 28.3 kPa (4.1 psi) is less than [ ]. Therefore, there is a low probability that a few of the 157 fuel assemblies in the AP1000 core could build up a debris bed that could exceed the acceptance criteria. However, many of the fuel assemblies in the core will have debris beds that have lower resistances than the acceptance criterion. The results of the statistical analysis of the AP1000 fuel assembly debris testing show that the effective core inlet adjusted dP would be [ ], using the 95 percent upper bound standard deviation, which demonstrates considerable margin to the acceptance criterion of 28.3 kPa (4.1 psi).

The staff notes that the statistical analysis is not required as a part of the GSI-191 evaluation, and the size of the test dataset is not sufficient to form the basis for a sound statistical analysis. However, the staff finds that the statistical analysis provides useful supporting evidence that the there is low probability that debris entering the core and debris bed buildup would degrade the core cooling margin to the point that the ECCS acceptance criteria were not met.

The staff evaluated the core cooling capability of the AP1000 core during long-term cooling based on the review of the long-term cooling analyses and fuel assembly debris loading head loss testing. The staff concludes the following:

- The design basis of 90 percent debris bypass (i.e., 90 percent of the design basis containment debris entering the core through submerged breaks unfiltered, bypassing the circulation screens), determined by the limiting DECL break, is a conservative assumption for the long-term cooling evaluation. The limiting DECL break was determined to be the worst case break that could allow the maximum amount of unfiltered debris to enter the reactor downcomer.

- The applicant conducted WCOBRA/TRAC long-term cooling sensitivity analyses using large nonmechanistic flow resistance at the core entrance to simulate debris-induced flow blockage to determine adequate core cooling. The limiting core inlet dP and flow results from Sensitivity Cases 3 and 10 were then used to develop acceptance criteria for the fuel assembly debris loading head loss testing.

- The applicant ran 39 fuel assembly tests to show that the worst case debris that would be expected in the AP1000 reactor during long-term cooling would not exceed the dP acceptance criteria. These tests showed that the design-basis AP1000 fibrous and particulate debris and chemical precipitates assumed to exist in the AP1000 containment do not induce a high enough head loss through the fuel assembly to reduce flow into the

core to less than the minimum required to provide adequate long-term cooling following a LOCA.

On the basis of its review, the staff finds that the evaluations performed by the applicant showed that, with the design-basis containment debris loading, adequate core cooling in the AP1000 can be maintained during the post-LOCA recirculation long-term cooling period.

6.2.1.8.2.8 Head Loss and Vortexing

The applicant conducted head loss testing using plant-specific debris loads and flow rates to demonstrate the adequacy of the containment recirculation and IRWST screens. The debris types considered were particulate, fiber, and chemical precipitates. WCAP-16914-P and WCAP-16914-NP document the methodology, assumptions, and results. The staff evaluated these documents considering guidance from Enclosure 1, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Strainer Head Loss and Vortexing," and Enclosure 3, "NRC Staff Review Guidance Regarding Generic Letter 2004-02 Closure in the Area of Plant-Specific Chemical Effect Evaluations," to "Revised Guidance for Review of Final Licensee Responses to Generic Letter 2004-02, 'Potential Impact of Debris Blockage on Emergency Recirculation during Design Basis Accidents at Pressurized-Water Reactors,'" dated March 28, 2008.

WCAP-16914-P documents all eight AP1000 screen head loss tests. These experiments were performed over a period of time, capturing different evolutions of the design. Because there were significant changes to the debris source term and screen size over the course of the program, only Tests WE213-4W and WE213-5W are representative of the final AP1000 design. These two tests, which are the focus of the subsequent discussion, were run with nearly identical parameters.

The AP1000 containment recirculation and IRWST screens comprise pockets constructed from perforated plate with holes less than or equal to 1.59 mm (0.0625 in). The screen used in the test is similarly constructed, [

].

The test facility included a large acrylic tank containing the test screen and supporting structure, positioned on the tank floor. Fluid entered the tank through a submerged sparger, located close to the upstream tank end. The flow entered the test screen pockets horizontally and was recirculated through piping connected to the downstream end of the tank. The pump, flow meter, and valves were external to the tank. The temperature of the fluid was maintained just below room temperature via cooling coils wrapped around the sparger.

Tests WE213-4W and WE213-5W began with clean screen head loss measurements at each of the two flow rates designated for use during the test to provide a reference point for the subsequent test results. The flume was then stabilized at the higher flow rate, and the particulate was added by slowly sprinkling the particulate surrogate at the water surface furthest upstream of the screen. The flume was run for five turnovers and stirred to resuspend any settled particulates. The fiber was then added in small batches. For each addition, a portion of the dry fiber was shaken into a solution of flume water and then slowly poured on the water

surface furthest upstream of the screen. When fiber addition was complete, the flume was stirred and then allowed to run at a steady state overnight. The next morning, the flow rate was reduced to the minimum value in preparation for chemical addition. The chemical precipitates, created outside the loop, were slowly added to the flume in small batches, at a rate that bounded twice the predicted rate of chemical production in the AP1000, scaled to the limiting screen surface area. Upon completion, the flume water was stirred and again allowed to run at a steady state overnight. The next day, the flow rate was increased to the maximum value and swept back before ending the test. For both Test WE213-4W and test WE213-5W, the head loss remained very close to zero throughout the entire test, from the initial clean screen step through to the final flow sweeps.

The particulate debris comprises the particulate portion of latent debris and the coatings in the ZOI, which fail as fine particles. Coatings that fail as chips are not transported to the screen, as previously discussed. The particulate surrogate was [ ], which is consistent with the description of "dirt" from Appendix V to NEI 04-07. The surrogate for the latent fiber was [ ], which is consistent with the recommendations in Appendix VII to NEI 04-07. It comprised heat-treated fiber that had been either shredded or chipped. For Tests WE213-4W and WE213-5W, the fiber was submersed in a bucket of water and thoroughly shaken. It was not added to the flume until it was confirmed that all fibers were sufficiently fine and individualized. This approach eliminated the fiber agglomeration that was observed in some of the earlier tests. [ ] was used as a surrogate for all three chemical products generated in the postaccident AP1000 sump. As discussed in Section 6.2.1.8.2.4 of this report, this is an appropriate surrogate. The chemical reactant products were formed outside the test loop in accordance with the NRC-approved methodology from WCAP-16530-NP-A. The debris sequencing in Tests WE213-4W and WE213-5W is consistent with the guidance. The flume water was stirred for 1 minute after each debris type was added, which is expected to be sufficient time for all debris to initially reach the screen. Visual confirmation was not possible because the particulates, which were added first, turned the water gray.

For fiber and particulate debris, the loadings for the IRWST and containment recirculation screen are found by dividing the amount of debris transported to each screen by the screen frontal area. The results, based on values taken from the DCD, are presented in Table 5-2 of WCAP-16914-P along with the calculated debris loading from Tests WE213-4W and WE213-5W. As shown, the test debris loadings encompass both the IRWST and containment recirculation screens. Scaling the tested amounts of debris by the more limiting IRWST frontal areas and transport assumptions demonstrates the testing bounds a containment that has [ ] of fiber and [ ] of particulate. The AP1000 design has 3.0 kg (6.6 lb) of fiber and 87.8 kg (193.4 lb) of particulates; thus, there is significant margin in the tested amount of particulates.

A different approach was taken to determine the amount of chemicals added to the test flume. The intended amount, shown in Table 5-2 of WCAP-16914-P, was determined by scaling the chemical debris to screen frontal areas. The applicant claimed this amount of chemicals would result in a chemical concentration in the test flume about [ ] times higher than expected in the AP1000 containment. In order to remove some of this conservatism, the applicant added a termination criterion to the test plan that would allow the test to conclude if the measured [ ]. For both Test WE213-4W and test WE213-5W, the flume was sampled after a portion of the chemical precipitates was added; [

].
The applicant provided additional information about the concentration measurement procedures in response to RAI-SRP6.2.2-CIB1-26 and RAI-SRP6.2.2-SPCV-31. The staff did not conclude that the [ ] provided a conservative alternative to area-based scaling. However, based on the amount of chemical debris added and the measured head loss, the staff did conclude that the test conformed to the staff's guidance with respect to the amount of chemical debris. This is discussed below with the test results.

The procedures for Tests WE213-4W and WE213-5W included minimum and maximum flow rates, originally meant to differentiate the higher IRWST injection flows from the lower recirculation flows. The fiber and particulates were added at the maximum flow rate, representative of injection flow, and the chemicals, which would not precipitate until after start of recirculation, were added at the minimum flow rate. Tests WE213-4W and WE213-5W specified the same maximum flow rate, but the minimum flow rate in Test WE213-5W was slightly higher than in Test WE213-4W in order to encompass the recirculation flow oscillations observed in the long-term cooling analysis, APP-PXS-GLR-001. The highest flows in the AP1000 occur during operation of the nonsafety-related RNS system, which was not intended to be bound in the original test plan. However, the pressure drops remained near zero throughout the testing, indicating a clean screen. This is consistent with photographs from WCAP-16914-P, which clearly show open areas on the screen. Additionally, the calculated fiber bed thickness for these tests is [ ], which is [ ] times thinner than what was demonstrated to produce a thin bed in one of the earlier tests.

As stated above, the staff determined that, although not all of the chemical debris was added to the test, the test conformed to the staff's guidance for test termination with respect to chemical debris. In Section 16 of the March 2008 guidance for GSI-191 chemical effects, the staff stated that tests should be terminated in a way that demonstrates with high confidence that additional time or chemical debris would not significantly change the maximum head loss. In Tests WE213-4W and WE213-5W, the applicant added [ ] percent of the prepared chemical debris, which was scaled to screen frontal area (Table 6-1). The screen area was clean and there was no measured head loss even after adding a substantial fraction of the debris. The staff concluded that the test termination conformed to the staff's guidance.

Because the pressure drop remained negligible during the flow sweeps, the applicant was able to demonstrate that the screen performance was not a function of flow rate and the DCD criterion was changed to require that the testing encompass RNS operation. The test report incorporated this change in Table 5-2, which identifies maximum AP1000 flow rates as those associated with RNS operation. As shown, the maximum tested flow rate bounds the containment recirculation screen, but it is only 94 percent of the IRWST scaled value. The applicant considers this acceptable because a 6-percent difference is small, the scaling was conservatively based on frontal area rather than surface area, and the test results demonstrate a clean screen. The staff agrees that the conservatism in the scaling bounds the minor difference in flow rates and that the resultant clean screen will be insensitive to a 6-percent higher flow rate.

The DCD-specified head loss limit of 1.7 kPa (0.25 psia) was derived from Sensitivity Case 3 of APP-PXS-GLR-001. In APP-PXS-GLR-001, Table 4-1, the containment recirculation screen flow was modeled by the PXS A flow at a temperature of 93.3 °C (200 °F), and the IRWST screen flow was modeled by the PXS B flow at a temperature of 60.6 °C (141 °F). The calculated head loss for the containment recirculation screen was [ ] at a flow

rate over the frontal area of [ ], and the calculated head loss for the IRWST screen was [ ] at a flow rate over the frontal area of [ ]. The long-term cooling analysis modeled the pressure drop as proportional to the value of flow squared. Applying this relationship at the minimum test flow from Test WE213-4W gave a pressure drop limit of [ ] at the containment recirculation screen and [ ] at the IRWST screen. The test plan conservatively set the head loss limit to 1.7 kPa (0.25 psia) at all flow rates, which is more restrictive than the calculated value for either screen. The head loss measured throughout the test was negligible, which demonstrates additional margin.

The flume water level was set about [ ] above the top of the screen, bounding both the containment recirculation screens, which have several feet of submergence, and the IRWST screens, which have a minimum submergence of 7.1 cm (2.8 in). As stated in Section 7.3 of WCAP-16914-P, no vortex formation or air entrainment into the screens was observed during any of the testing. [

]. This experiment is considered bounding as the current design demonstrates a clean screen. The relatively small submergence of the IRWST screens could lead to concerns regarding flashing and deaeration if the pressure drop across the screen approached the head loss limit. However, because the testing demonstrated a clean screen and no vortexing was seen, even during the flow sweeps that encompass RNS operation, there is no concern of voiding or potential entrainment of vapor if potentially saturated liquid passes through the screens. There is also no concern that additional water could be held up in the IRWST as the water level drops below the screen during the wall-to-wall flooding case described in DCD Section 15.6.5.4C.3.

6.2.1.8.2.9 Downstream Effects—In Vessel

During the post-LOCA containment recirculation long-term core cooling phase, the containment debris and chemicals enter the reactor vessel, causing potential core flow blockage, boron precipitation, and plateout of chemical precipitates on the fuel cladding, resulting in degradation of core heat transfer. Section 6 of TR-26, presents an in-vessel evaluation, which assesses the impact of debris in the post-LOCA recirculating water on components inside of the reactor vessel, including the core inlet and fuel assemblies.

The effects of debris in the post-LOCA recirculating water flowing through the fuel assemblies were evaluated through the long-term cooling sensitivity analysis and the fuel assembly head loss testing, as discussed in Section 6.2.1.8.2.7 of this SE. This section provides an evaluation of the effect of potential chemical deposition and scale buildup on the fuel rods on maintaining effective heat transfer from the fuel rods to the coolant.

During post-LOCA long-term cooling, the AP1000 PXS design uses ADS-4 valves connected to the HLs to vent steam and a considerable amount of water from the RCS. The water that leaves through the ADS-4 valves carries boron and other chemicals out of the RCS, which automatically and effectively limits the buildup of these chemicals in the core. Therefore, boron precipitation in the reactor vessel is prevented by sufficient flow of PXS water through the ADS-4 valves to limit the increase in boron concentration of the water remaining in the reactor vessel. DCD Section 15.6.5.4C.4 describes an evaluation of post-LOCA long-term core cooling, as well as core boron concentration and boron precipitation. The analysis results indicated that the ADS-4 venting quality at the initiation of recirculation is about 50 percent and decreases to

less than 10 percent at the time of wall-to-wall flooding. At the maximum vent quality, the maximum boron concentration peaks at about 7,400 ppm at the time of recirculation initiation. After this time, the core boron concentration decreases as the ADS-4 vent quality decreases, reaching 5,000 ppm about 9 hours after the accident. The maximum boron solubility temperature is 14.4 °C (58 °F) at 7,400 ppm, which is virtually unattainable in the reactor vessel or the ADS-4 vent pipe. For the containment floodup case, the minimum sump injection head is adequate to maintain core cooling and limit boron concentration. The results show that venting of steam and water ensures that there is adequate liquid flow through the core to cool it and to prevent boron precipitation. Section 15.2.7 of NUREG-1793 discussed the staff's evaluation of the boron precipitation and concluded that: (1) the core remains cooled for the duration of the long-term cooling phase; (2) the boron concentration in the core keeps the core noncritical; and (3) boron precipitation will not occur to obstruct core coolant flow.

However, the long-term cooling analysis presented in the DCD was performed without consideration of containment debris. Because of the flow conditions of the core where there is a significant blockage at the core inlet created by unfiltered debris through the break, The applicant performed a long-term core cooling sensitivity study to evaluate the effects of core inlet blockage. This sensitivity study is described in APP-PXS-GLR-001, Revision 4.

The AP1000 long-term cooling sensitivity study results show that as the core inlet flow resistance increases, the core flow rate decreases, the quality of flow discharged through ADS-4 valves increases, and the boric acid concentration increases. For Sensitivity Case 10, which has the highest core inlet flow resistance, the core flow rate is predicted to be 65 lbm/s with a bounding pressure drop of 4.1 psid, the ADS-4 discharge flow quality is 0.49, and the maximum concentration of boron is 6,100 ppm in the core. This is less than the maximum concentration of 7,400 ppm in the DCD evaluation. Therefore, there is no concern with precipitation in the lower plenum during long-term cooling following a LOCA since the reactor water is not cooled to temperatures lower than the corresponding boron solubility temperature during long-term cooling.

In RAI-SRP6.2.2-SRSB-26, the staff asked if there were any situations where two-phase flow behavior could challenge the single-phase fuel assembly debris-induced head loss test results, and whether a different liquid temperature or local boiling phenomenon affects the behavior of the debris plugging the core. In partial response to RAI-SRP6.2.2-SRSB-26, the applicant provided TR APP-GW-GLR-110, Revision 0, which provides an evaluation of the potential for plateout of unbuffered boric acid or buffered boric acid on the fuel rod surface. The applicant addressed the effect of boron plateout on the core for high steam qualities in the core region that may occur during a significant core inlet blockage by debris. The applicant had previously conducted a series of single-rod bench-scale tests to investigate the nucleate boiling heat transfer characteristics of unbuffered and buffered boric acid solutions. These tests included concentrations of boric acid and buffer agent TSP that were equal to and greater than those concentrations that will occur in the AP1000 following a LOCA. The applicant also reviewed additional PWR heated rod testing in the presence of boric acid solution with decay heat level heat input and low pressure for application to the AP1000. These additional heated rod tests included rod-bundle geometries (Tuunanen, J., et al., "Experimental and Analytical Studies of Boric Acid Concentrations in a VVER-440 Reactor during the Long-Term Cooling Period of Loss of Coolant Accidents," *Nuclear Engineering and Design*, issued 1994) and multirod full-height slab core geometry (W3F1-2005-0007, "Supplement to Amendment Request NPF-38-249, Extended Power Uprate," dated February 5, 2005). These more prototypical geometries of the multirod and rod bundle test displayed precipitation behavior in the heated rod region that was consistent with the single heated rod testing for unbuffered boric acid and boric acid buffered

with TSP. These tests generally showed no bulk precipitation in the heated rod region of the core and some local precipitation in the boiling region if the core would become uncovered. The form of boric acid precipitation is usually amorphous and can be redissolved in the presence of a continuous liquid phase. Therefore, the applicant does not expect the deposition of boric acid or boric acid buffered with TSP to occur during post-LOCA conditions in the AP1000 since the heated core region has been shown to be covered at all times by a two-phase mixture.

For the AP1000 design, the boron concentration calculated for the limiting long-term cooling Sensitivity Case 10 is only 6,100 ppm, which has a corresponding low solubility temperature that is not attainable during the long-term core cooling period. Since the available test data demonstrate that, if the boron precipitation occurs at high boron concentration, boron precipitation on the fuel is generally amorphous and would be redissolved in the presence of continuous liquid phase, the staff agrees that potential boric acid solute plateout on the fuel cladding is not likely, and boron precipitation is not a concern for the AP1000 design.

To address the concern that post-LOCA containment debris and chemical precipitates can plate out on fuel rod cladding and impede the heat removal from the fuel rods, The applicant evaluated the impact of post-LOCA deposition of chemical precipitates on fuel rods for the AP1000 design. The source of chemical products is the interaction of the fluid inventory in the post-LOCA containment sump environment with debris and other materials exposed to and submerged in the sump fluid. The purpose of the evaluation is to predict a maximum scale thickness of the resulting cladding deposit buildup and the maximum clad/oxide interface temperature resulting from the deposits during the post-LOCA recirculation long-term core cooling phase.

The evaluation, described in Section 6.2.2 of TR-26, Revision 8, uses the Loss-of-Coolant Accident Deposition Model (LOCADM). LOCADM is a spreadsheet calculation tool developed by the PWR Owners Group to conservatively predict chemical interaction precipitate formation and buildup of chemical deposits on fuel cladding after a LOCA. The details of the LOCADM analysis model appear in Section 7 and Appendix E of WCAP-16793-NP, "Evaluation of Long-Term Cooling Considering Particulate, Fibrous and Chemical Debris in the Recirculating Fluid," Revision 1, issued April 2009.

WCAP-16793-NP, Revision 1, also proposes the following two acceptance criteria for the post-LOCA evaluation of the chemical deposition on the fuel rods, which the staff accepted for GSI-191 consideration:

(1) The maximum cladding temperature during recirculation from the containment sump will not exceed 426.7 °C (800 °F).

(2) The thickness of the deposition of debris, chemical precipitates, or both on the cladding (oxide + crud + precipitate) will not exceed 50 mils (1270 Micrometer (μm)).

Evaluation of the LOCADM Analysis Model

LOCADM predicts both the deposit thickness and cladding surface temperature as a function of time at a number of core locations. Although the LOCADM analysis model is described in WCAP-16793-NP, Revision 1, there is a link to WCAP-16530-NP-A since the chemical model contained in TR WCAP-16530-NP-A is used to develop the potential source term of species that may enter the reactor vessel. WCAP-16530-NP-A provides a method for evaluating plant-specific chemical effects in a post-LOCA environment, including guidance for how to

prepare surrogate chemical precipitates that may be used in strainer head loss tests. The staff reviewed and approved WCAP-16530-NP. WCAP-16530-NP-A, however, does not explicitly address potential chemical effects that may occur in the reactor vessel. WCAP-16793-NP, Revision 1, evaluates potential chemical effects that may occur downstream of the sump strainer in the reactor vessel. The materials tested in WCAP-16530-NP included:

1. commercially pure aluminum and galvanized steel,
2. calcium silicate (Cal-Sil) insulation,
3. NUKON™ fiberglass,
4. other fiberglass - Temp Mat™,
5. Interam™ E-class insulation,
6. powdered concrete,
7. mineral wool insulation,
8. microporous insulation (e.g., Min-K™), and
9. fire-retardant material (e.g., FiberFrax™).

WCAP-16530-NP describes a number of dissolution tests conducted to examine the chemical behavior of various materials found in the sump environment. Sampling times for the dissolution test were set at 30 minutes, 60 minutes, and 90 minutes. The results of the WCAP-16530 test program are consistent with previous work such as the integrated chemical effect test (ICET) program and show that:

1) The predominant materials leached from containment materials are:

   - aluminum ions
   - silicates
   - calcium ions

2) The predominant chemical precipitates formed are:

   - aluminum (oxy) hydroxide
   - sodium aluminum silicate
   - calcium phosphate (for plants using TSP for pH control)

It is possible that other silicate materials may be generated (e.g., calcium aluminum silicate or zinc silicate), but their contribution, based on the referenced studies, will be small (contributing less than 5 percent of the total mass) relative to the predominant precipitates.

The WCAP-16530-NP model considers the release rates of aluminum, calcium and silicate, as these provide the greatest masses of materials that can become insoluble and impacts of other materials are negligible. Given a source term of material from the WCAP-16530-NP model, the staff reviewed the methodology used to determine that these materials:

1. Would not deposit on fuel surfaces to the extent that heat transfer is unacceptably low, and

2. Would not block flow through the fuel channels should the scale materials deposited become dislodged by spalling during fuel cool down.

In evaluating the potential for plate-out of dissolved or suspended chemical compounds on the fuel surface, the WCAP-16793-NP methodology assumes that all of the dissolved species and compounds resulting from the WCAP-16530-NP assessment are transported through the containment sump screen to the reactor vessel. This material represents the source term in WCAP-16793-NP for evaluating plate-out of scale-forming materials on the fuel cladding. The staff finds this source term assumption to be acceptable since the chemical source term is based on WCAP-16530-NP testing, and it is conservative for the reactor vessel fuel analysis to assume that no precipitate settles on the containment floor, no precipitate becomes trapped in a filtering debris bed covering the sump strainer, and material does not deposit in other locations downstream of the strainer (e.g., heat exchangers, reactor vessel lower plenum).

Although the staff finds the use of the chemical model spreadsheet in WCAP-16530-NP to be acceptable for determining the chemical source term for LOCADM, a limitation and condition was provided in the safety evaluation for WCAP-16530-NP related to the aluminum release rate. The WCAP-16530-NP chemical model aluminum release rate is based, in part, on a fit to ICET data using an averaged 30-day release. Actual corrosion of aluminum coupons during the ICET test appeared to occur in two stages; active corrosion for the first half of the test followed by passivation of the aluminum during the second half of the test. Therefore, while the 30-day fit to the ICET data is reasonable, the WCAP-16530-NP model under-predicts aluminum release by about a factor of 2 during the active corrosion part of ICET 1. This is important since the in-core LOCADM chemical deposition rates can be much greater during the initial period following a LOCA, if local conditions predict boiling. To account for potentially greater amounts of aluminum during the initial days following a LOCA, a user's LOCADM input shall apply a 2x increase to the WCAP-16530 spreadsheet predicted aluminum release, not to exceed the total amount of aluminum predicted by the WCAP-16530-NP spreadsheet for 30 days. In other words, the total amount of aluminum released equals that predicted by the WCAP-16530-NP spreadsheet, but the timing of the release is accelerated. Alternately, users may choose to use a different method for determining aluminum release, but users shall not use an aluminum release rate equation that under-predicts the aluminum concentrations measured during the initial 15 days of ICET 1. If a user uses plant-specific refinements to reduce the chemical source term calculated by the WCAP-16530-NP base model, the user shall provide technical justification demonstrating that the refined chemical source term adequately bounds the postulated plant chemical product generation.

WCAP-16793-NP uses various heat transfer computer programs [ANSYS Mechanical Software and WCOBRA/TRAC WCAP-12945-P-A] and a commercially available calculational software package (MATHCAD) for estimating the effects of the plateout of dissolved materials on the increase in fuel clad temperature. WCAP-16793-NP relied on the LOCADM code for its final assessments since the LOCADM calculations address non-uniform chemical deposition due to variation of core power and boiling.

The starting assumption for the LOCADM model with respect to chemical effects is that all the dissolved and suspended chemicals pass through the containment sump screen and into the reactor core. This is a conservative assumption because it maximizes the amount of chemicals available to cause deleterious effects.

The LOCADM model also assumes that some of the fibrous material from destroyed insulation is not removed by the sump strainer and that this material also passes on to the reactor core area. The mass of fiber passing through the strainer is determined on a plant-specific basis, based on bypass testing. LOCADM assumes instantaneous chemical participation of this fiber. Therefore, the fiber bypass quantity is converted to a mass of fiberglass and then to an

equivalent mass of elements [calcium + aluminum + silicon] that is immediately available to be deposited in the LOCADM analysis. This increase in the mass of dissolved chemicals is compared to the original mass of dissolved chemicals determined by the WCAP-16530-NP calculations [calcium + aluminum + silicon] and a percent increase is calculated. This increase is on the order of one to two percent, and is referred to as a "Bump-Up Factor."

These two chemical sources are then used in the plant-specific application of LOCADM. Given the potential plant-specific chemical source term in the reactor vessel, LOCADM determines the amount of scale that deposits on the fuel over time and then calculates maximum fuel clad temperature. An assumption that is very important to the LOCADM calculations is the coefficient of thermal conductivity for the chemical deposits. In order to determine an appropriate thermal conductivity coefficient for the LOCADM calculations, two different thermodynamic equilibrium based codes were used to assess the chemical species that may form in the post-LOCA reactor vessel environment. The applicant performed an analysis using the HSC program by Outokomptu. This thermodynamic equilibrium code was used to evaluate potential differences in the predicted species and to support the choice of a limiting thermal conductivity value for a chemical deposit that may form on the fuel. Using the chemical species predicted by these thermodynamic equilibrium analyses, a lower-bound thermal conductivity value was selected for the LOCADM analysis in WCAP-16793-NP to minimize heat transfer and maximize the temperature rise on the fuel surfaces. A chemical deposit thermal conductivity value of 0.19 watts per meter-degree Celsius (W/m-ºC) (0.11 British thermal unit per foot-hour-degree Fahrenheit (BTU/ft-hr-°F)) was selected based on the possible formation of a postulated sodium aluminum silicate scale. A thermal conductivity value of 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) is the minimum thermal conductivity value reported for sodium aluminum silicate scale. For comparison, the thermal conductivity of dry fiberglass insulation is approximately 8.65×10^{-2} W/m-ºC (0.05 BTU/ft-hr-°F), and, with eight percent of its mass wetted, it increases to approximately 0.173 W/m-ºC (0.1 BTU/ft-hr-°F). The staff questioned if there were any materials from the thermodynamic predictions for fuel clad surface deposits which could have lower thermal conductivity values. The applicant responded that 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) was a bounding thermal conductivity value reported for any of the postulated species that could form a scale deposit on the fuel clad surface. Information provided by the applicant in RAI response number 34 [Schiffley, F. P., PWROG letter to Document Control Desk, NRC, "Response to the NRC for clarification to Requests for Additional Information (RAI) on WCAP-16793-NP," showed thermal conductivity coefficients of representative calcium-based boiler scale deposits that were in the 0.52 to 0.865 W/m-ºC (0.3 to 0.5 BTU/ft-hr-°F) range, and the thermal conductivity of glass was reported as 1.02 W/m-ºC (0.59 BTU/ft-hr-°F).

Since the LOCADM calculations do not consider the presence of large debris, the staff questioned whether small pieces of insulation ("fines") incorporated into a deposit could result in a lower thermal conductivity value than the 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) assumed for a sodium aluminum silicate scale. The applicant responded that since core temperatures have decreased by the time the ECCS switches from injection to recirculation mode, which is the time when the first fibrous debris could bypass the sump screens and enter the core, the temperature of the core is insufficient to cause melting of the fiberglass or other fibrous material. Therefore, the presence of fiber fines would not create a different type of scale other than that predicted by the thermodynamic models. The applicant also responded that although dry fiberglass has a lower thermal conductivity than the 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) assumed for the chemical deposit, a fiber deposit would be porous and would allow water to fill in the porosity. Since water has a much higher thermal conductivity than air, the overall thermal conductivity for a deposit containing fiberglass would be bounded by the assumed 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) value. This reasoning is supported by literature [Joint Departments of the

Army and Air Force, USA, Technical Manual TM 5-852-51/AFR 88-19, Volume 5, Article Sub-Arctic Construction: Utilities, Chapter 12] that indicated the fiberglass thermal conductivity constant increases by a factor of two with an eight percent volume of water incorporated into its structure. This is also consistent with insulation manufacturer recommendations to change insulation if it is wetted since the heat conduction through the insulation increases; in other words, it is no longer an effective insulator.

Based on the above discussion, the staff finds that the 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) thermal conductivity value assumed for deposition of scale and particulate represents an acceptably low value to help achieve a conservative prediction of fuel clad temperature increases due to chemical deposits. If plant-specific calculations use a less conservative thermal conductivity value for scale, i.e., greater than 0.19 W/m-ºC (0.11 BTU/ft-hr-°F), the staff expects the licensee to provide a technical justification for the plant-specific thermal conductivity to the staff. This justification should demonstrate why it is not possible to form a sodium aluminum silicate scale or other scales with conductivities below the selected plant value.

Given the potential chemical source term and using a conservative value for thermal conductivity, LOCADM calculates deposit growth over time. The default initial oxide and crud thicknesses assumed by LOCADM are based on the fuel age and the limiting values that have been measured at modern PWRs. Since the boiling deposition mechanism results in the most rapid deposit growth and forms the most tenacious deposits, LOCADM assumes that all deposition occurs through the boiling process if conditions at a core node predict any boiling. The amount of scale calculated to be deposited under boiling assumes that 50 percent of the water present at the clad surface boils and all solutes transported into the deposit by boiling are deposited locally, as liquid evaporates, at a rate proportional to the steaming rate. Subsequent plate-out of solids, once boiling subsides, is estimated from other literature sources (WCAP-16793-NP, Revision 1, RAI Set #2, RAI #8) to be 1/80th of the solids deposition rate during boiling based on the temperatures encountered at the fuel. Once formed, deposits are assumed not to thin by flow attrition, dissolution, or spalling. The sample LOCADM calculation in WCAP-16793-NP, Revision 1, included a 3188 megawatt-thermal PWR with high fiber (198 m^3 (7000 ft^3)) and a large quantity of calcium silicate insulation (2.27 m^3 (80 ft^3)). The staff questioned what additional effect the existing clad crud film and oxide scale (from three cycles) would have on the LOCADM calculations. The applicant responded that the sample LOCADM calculation, for the conditions stated above, including initial fuel clad oxide and crud, showed the maximum chemical scale thickness calculated over 30 days was 10 mils (0.010 in). The maximum clad surface temperature after the start of recirculation was 162.2 °C (324 °F), which meets the acceptance criteria of 426.7 °C (800 °F).

Since LOCADM does not directly account for fiber fines bypassing the sump screen, the staff also questioned how possible effects from fibers depositing in the core are assessed. Analysis of core inlet blockage is discussed elsewhere in this report, but modeling demonstrated that with 99 percent of the core flow blocked, sufficient cooling water would be provided as a result of boiling and back flow from above to prevent clad temperatures exceeding 426.7 °C (800 °F). To model potential local hot spots, heat transfer analysis was provided in Appendix D of WCAP-16793-NP, Revision 1, assuming heat transfer in the radial direction only (i.e., ignoring any axial heat transfer) and using a chemical scale thermal conductivity of 0.173 W/m-ºC (0.1 BTU/ft-hr-°F). These calculations showed that for a chemical scale thickness of 0.127 cm (50 mils or 0.050 in) that formed "instantaneously" at the start of recirculation, the maximum fuel clad surface temperature for a fuel rod diameter of 0.91 cm (0.36 in) is 293.3 °C (560 °F). Additional analyses were performed for larger diameter fuel rods, 1.06 cm (0.416 in) and 1.07 cm (0.422 in) OD rods. The predicted peak clad-oxide interface temperature was less than the

acceptance basis value of 426.7 °C (800 °F) in each case. The staff finds this analysis to be acceptable since the assumptions of instantaneous chemical precipitate formation, heat transfer only in the horizontal plane (radial direction), and the assumed thermal conductivity for chemical scale are judged to be conservative for reasons stated in the WCAP.

The staff also questioned whether blockage of core flow channels might occur from scale initially deposited on the fuel surface that would flake off during the cool down process. The applicant responded that the thickness of the scale formed is limited by the amount of solids dissolved in the water. Using scale deposition models the applicant demonstrated that the thickest scale fragment would be insufficient to bridge a fuel rod to fuel rod span to block flow. The staff finds this justification acceptable because the spalling process from the fuel is slow, and experience from spent fuel pool debris generated at PWRs shows these scale materials to be granular and of small size rather than large flakes.

The staff reviewed the LOCADM analysis model as described in WCAP-16793-NP, Revision 1, and finds that:

1. The mass of material used to determine the debris and scale loading is conservative based on the source term calculated from the WCAP-16530-NP tests, along with the assumption that no precipitates settle on the containment floor, are filtered at the sump screen, or deposit in heat exchangers, piping, or in the reactor vessel outside of the core. The mass of materials includes a “bump up factor” to account for fibrous material that bypasses the sump screens. The staff finds this bump up factor to be acceptable for reasons stated in this SER section.

2. The thermal conductivity assumed for chemical scale and debris deposits represents an acceptably low value (0.19 W/m-ºC (0.11 BTU/ft-hr-°F) to help achieve a conservative prediction of fuel clad temperature increase. Wetted insulation allows for better conduction of heat and the thermal conductivity of wetted insulation would be higher. Thus the use of 0.19 W/m-ºC (0.11 BTU/ft-hr-°F) is a conservative assumption.

3. Industry-recognized calculation models were used to predict temperature increases at the fuel surface as a result of chemical plate-out, and these models confirm that the limit of 426.7 °C (800 °F) is not exceeded when these models are used in conjunction with the source term assumptions in WCAP-16530.

4. Blockage of fuel rod spans by spalled fuel scales is unlikely due to the time dependency for spalling and the small thickness of the scale compared to the space between the fuel rods.

Based on the above, the staff concludes that the LOCADM analysis model provides a valid approach to determining potential flow restrictions due to chemical effects of RCS liquid and containment debris and materials, and is both conservative and representative of the post-LOCA conditions based on chemical reactions described in WCAP-16530-NP. Therefore, given the acceptance criteria for fiber bypass, the staff concludes the chemical effects on core cooling resulting from debris and scale deposition following a LOCA are insufficient to create a condition resulting in fuel clad temperatures exceeding the temperature limit of 426.7 °C (800 °F). However, the acceptability of the application of the LOCADM analysis model is contingent upon the following conditions.

1. The aluminum release rate equation used in WCAP-16530-NP provides a reasonable fit to the total aluminum release for the 30-day ICET tests but under-predicts the aluminum concentrations during the initial active corrosion portion of the test. To provide more appropriate levels of aluminum for the LOCADM analysis in the initial days following a LOCA, users shall apply a factor of two to the aluminum release rate as determined by the WCAP-16530-NP spreadsheet. If a user chooses to use a different method for determining the aluminum release, it must demonstrate that the method does not under-predict the aluminum concentrations measured during the initial 15 days of ICET 1.

2. If plant-specific refinements are made to the LOCADM base model to reduce conservatisms, the user shall demonstrate that the results still adequately bound chemical product generation. If a user uses plant-specific refinements to the WCAP-16530-NP-A base model that reduces the chemical source term considered in the downstream analysis, the user shall provide a technical justification that demonstrates that the refined chemical source term adequately bounds chemical product generation. This will provide the basis that the reactor vessel deposition calculations are also bounding.

WCAP-16793-NP, Revision 1, states that the material with the highest insulating value that could deposit from post-LOCA coolant impurities would be sodium aluminum silicate. The WCAP recommends that a thermal conductivity of 0.19 W/m-°C (0.11 BTU/ft-hr-°F) be used for the sodium aluminum silicate scale and for bounding calculations when there is uncertainty in the type of scale that may form.

To demonstrate acceptable AP1000 long-term core cooling performance, the applicant performed an evaluation using the LOCADM spreadsheet to account for chemical reactions within the coolant that could lead to deposition of material within the core. This evaluation was documented in Section 6.2.2 of TR-26 and Westinghouse calculation note APP-PXS-M3C-057, "Loss of Coolant Accident Deposition Model (LOCADM) Analysis for AP1000 Plant Design," Revision 1, issued November 2009. The AP1000 LOCADM evaluation makes the following assumptions and simplifications:

- AP1000 Unique Design Features:

  In the AP1000 design, the containment spray system is locked out during a LOCA. Therefore, the containment spray is not considered in the calculation of the post-LOCA debris source release.

  The AP1000 plant design relies on the ADS-4 valves in the HL to vent significant quantities of water along with steam from the core to the containment throughout the LOCA event. This behavior is modeled in the LOCADM spreadsheet by defining core injection flow rates that exceeded the boiloff rate by an amount calculated with the decay heat.

- Treatment of Aluminum:

  Although the AP1000 design precludes a large amount of aluminum from making contact with post-LOCA containment fluids, a mass of 27.2 kg (60 lb) of aluminum is assumed for conservatism. The aluminum surface area is increased to account for the zinc release from galvanized steel, which is not an input for LOCADM. Increasing the

aluminum surface area is conservative because the aluminum release rate is greater than that of any other material used in this evaluation.

The aluminum release rate is modified to satisfy NRC concerns about the trend of the predicted aluminum corrosion, by doubling the release rate during the initial portion of the event, yet it holds fixed the total aluminum mass release. This is consistent with the condition of WCAP-16530-NP. It is important because the release rate of aluminum is increased early in the transient when the deposition on the fuel is greatest because of high core decay heat rates and the boiling associated with the removal of that decay heat.

- Use of the Prefilled Reactor and Sump Option

  The LOCADM analysis assumes that the entire sump volume is present in the sump at time 0, precluding the need to specify individual break flow rates. This is conservative, as the entire sump volume is immediately available to react with the submerged debris at the start of the transient, and provides for the calculation of a greater amount of chemical precipitate deposition on the fuel.

- Core deposition is assumed to begin at the start of recirculation (9,300 seconds) and continue for the 30 days evaluated.

- Use of the Bump-Up Factor To Account for Fibrous Debris

  The bump-up factor used in LOCADM accounts for the postulated bypass of latent fibrous debris by increasing the mass of chemical precipitates that may be deposited on the fuel. To implement the bump-up factor in LOCADM, all materials that contribute to the formation of chemical precipitates are increased by a uniform percentage so that the resulting precipitates available for deposition have increased by approximately the amount of latent fibrous debris assumed for the AP1000. This method is independent of the type, diameter, or length of the fiber.

  The bump-up factor as applied to the AP1000 LOCADM evaluation is conservative because it is calculated (in APP-PXS-M3C-053, "AP1000 Latent Debris Calculation," Revision 2, issued November 2009) based on a fibrous debris loading of [ ]. This is the same debris loading used in the ex-vessel downstream effects analysis to evaluate the effects of debris in the recirculating water on pumps, valves, and other components in the post-LOCA recirculation flowpaths. This value is much higher than the design-basis containment residual fiber debris value of 3.0 kg (6.6 lb) and 90 percent transport to the reactor vessel. There are other conservatisms as discussed in TR-26, such as the low value of thermal conductivity (0.19 W/m-°C (0.11 BTU/ft-hr-°F) assumed in LOCADM for the scale buildup on the fuel rods when considering the latent fiber as part of the fuel rod post-LOCA scale.

  TR-26 indicates that the thermal conductivity of manmade fibers such as nylon and polyester (0.249 and 0.225 W/m-°C (0.144 and 0.13 BTU/ft-hr-°F) are higher than the assumed thermal conductivity for the scale. The thermal conductivity of natural fiber, such as cotton (3.46E-2 W/m-°C (0.02 BTU/ft-hr-°F) may be lower, but it will increase significantly when saturated with water, as is the case in a post-LOCA environment. The thermal conductivity of these saturated fibers rises significantly, trending towards the value of water at the ambient conditions saturating the fibrous material (~0.69 W/m-°C

(0.40 BTU/ft-hr-°F), which is much higher than the heat conductivity used for the chemical scale in the LOCADM evaluation.

TR-26 presented three scenarios evaluated with the LOCADM spreadsheet for the AP1000 design: (1) minimum sump volume case; (2) maximum sump volume case; and (3) minimum sump volume case with a bump-up factor for fiber deposition. The detailed evaluation described in Westinghouse calculation note APP-PXS-M3C-057, Revision 1, includes more sensitivity cases. The results of sensitivity analysis indicate that the minimum sump water volume results in a higher concentration of AP1000 accident chemical products and is therefore more limiting for the chemical deposition evaluation. All three cases described in TR-26 include the doubling of the aluminum release rate as recommended by WCAP-16793, Revision 1. The results show the post-LOCA scale thicknesses of [ ], respectively, for the three cases. With the preaccident oxide thickness of 0.15 mm (5.98 mils) and a crud thickness of 0.14 mm (5.51 mils), the total deposition thicknesses are [ ], respectively. These predicted thicknesses are significantly below the acceptance criterion of 1.27 mm (50 mils). The maximum temperature calculated for the outside diameter of the fuel cladding (at the fuel/oxide interface) is [ ] for all three cases, which is much less than the acceptance value of 426.7 °C (800 °F). This peak cladding temperature occurs at the onset of recirculation before significant debris deposition on the fuel cladding occurs. The chemical deposition appears to have an insignificant effect on the peak cladding temperature because the decay heat is decreasing faster than the chemical deposition rate.

In summary, the LOCADM calculations performed for the AP1000 demonstrate that both acceptance criteria for long-term core cooling identified previously in this report are achieved with significant margin. Specifically, the following is true for the cases evaluated:

- The maximum clad temperature calculated for the AP1000 of [ ] is significantly less than the acceptance value of 426.7 °C (800 °F).

- The total thickness of deposition calculated for the AP1000 fuel cladding is significantly less than the acceptance value for thickness of 0.127 cm (50 mils).

Therefore, the staff concludes that the AP1000 long-term core cooling capability remains viable in the presence of chemical deposition on the fuel cladding.

6.2.1.8.2.10 Debris Source Term

This section evaluates how the plant demonstrates and controls the debris source term. For the AP1000, this includes ITAAC, COL items, and technical specification (TS) surveillance requirements.

ITAAC related to the debris source term are included in DCD Tier 1, Table 2.2.3-4, as part of the 8c) design commitment that the PCS provide safety injection during design-basis events. ITAAC Item ix) verifies by inspection that insulation inside containment within the ZOI is MRI or that a report exists demonstrating that it is a suitable equivalent insulation. ITAAC Item ix) also verifies by inspection that other insulation inside the containment below the maximum DBA flood level is MRI, jacketed fiberglass, or a suitable equivalent insulation. Item x) verifies by inspection that reports exist concluding that tags and signs inside containment that are not inside cabinets or other enclosures have a density greater than or equal to 1.6 g/cm^3 (100 lbm/ft^3) and that ventilation filters and fiber barriers inside containment within the ZOI or below the maximum DBA flood level have a density greater or equal to 1.6 g/cm^3 (100 lbm/ft^3).

The staff finds these ITAAC acceptable because they capture key assumptions made in the long-term cooling analysis regarding debris transport and the amount of fibrous debris and because DCD Tier 2, Section 6.3.2.2.7.1, clearly defines the ZOI, the maximum DBA flood level, and the requirements for a suitably equivalent insulation.

COL Information Item 6.3-1 requires applicants referencing the AP1000 design to develop a cleanliness program that limits the debris left inside the containment following refueling and maintenance outages. Specifically, the amount of latent debris located within the containment must be less than 59.0 kg (130 lb) total latent debris, of which up to 3.0 kg (6.6 lb) is fibrous, and any outage materials stored inside the containment must not produce physical or chemical debris that could be transported to any of the filtering locations. The staff finds this acceptable because it is consistent with the recommendations in RG 1.82, Section C.1.1.2.1, which states that plant procedures should be established to regularly clean the containment and to control and remove foreign materials, and Section C.1.3.2.5, which states that the cleanliness program should be correlated to the amount of debris used in the long-term cooling analysis.

TS Surveillance Requirement 3.5.6.8 requires visual inspection of the IRWST and recirculation screens every 24 months to ensure that they are not restricted by debris. TS Surveillance Requirement 3.5.4.7 requires a similar 24-month inspection of the IRWST gutters. This is consistent with the long-term cooling analysis, which assumes the screens are clean before the LOCA.

6.2.1.8.2.11 Screen Design

ITAAC related to the screen design are included in DCD Tier 1, Table 2.2-3-4, as part of the 8c) design commitment that the PCS provide safety injection during design-basis events. Items vii) and viii) verify by inspection the key design features of the debris screens and barriers, including the existence of plates and debris curbs for the containment recirculation screen and the raised elevation of the IRWST screens. The screen frontal areas, surface areas, and mesh hole sizes are also verified to meet specific design criteria. Because these criteria are consistent with the head loss testing, they demonstrate that the as-built design will perform as expected.

Regulatory Position C.1.1.1.2 of RG 1.82, Revision 3 states that to the extent practical, the sumps should be physically separated from each other and from high energy piping systems by structural barriers to preclude damage by whipping pipes or high velocity jets of water or steam.

While there is physical separation between the IRWST screens, the applicant states that it was necessary to position the containment recirculation screens next to each other due to the location of the PXS subcompartment and the large size of the screens. To address pipe rupture and jet impingement vulnerabilities, the applicant has committed to demonstrate that the containment recirculation screens are protected by pipe whip restraints from the dynamic effects of pipe breaks in DCD Tier 2 Section 3.6.4.1 and DCD Tier 1 Table 3.3-6, Item 8). The staff finds that these restraints will protect the containment recirculation screens such that there are no credible pipe ruptures or jet impingement scenarios capable of causing screen failure. Therefore, this design meets the requirements of GDC 35.

Regulatory Position C.1.1.1.1 of RG 1.82, Revision 3 states that a minimum of two sumps should be provided, each with sufficient capacity to serve one of the redundant emergency core cooling lines. In the AP1000 design, while each PXS subsystem is associated with its own containment recirculation screen, the screens are cross connected. Likewise, each PXS subsystem is associated with its own IRWST screen, which is cross connected to a third IRWST

screen. Therefore, if one PXS subsystem does not draw water both containment recirculation screens or all three IRWST screens, will be available to support the functioning subsystem. The NRC recommended providing two sumps to establish clear compliance with GDC 35, which requires long-term mitigation capability of a loss of coolant accident assuming a single failure. However, because the screens are either protected by pipe whip restraints (containment recirculation screens) or not identified as essential targets for the dynamic effects of pipe breaks (IRWST screens), and because the screens and their anchorage are designed to withstand seismic loads and postaccident operating loads, including head loss and debris weight, the staff finds there is no credible chance of failure of the screens, and the design meets the requirements of GDC 35.

NUREG-1793, Section 3.9.3, includes an evaluation of the structural adequacy of the sump screens using guidance from RG 1.82.

6.2.1.8.2.12 Upstream Effects

Any potential effects that debris may have in transit from its source to either the IRWST screens or containment recirculation screens are termed "upstream effects." An evaluation of upstream effects ensures that flow necessary for recirculation is not held up by debris blockage at drains or other narrow pathways.

The applicant discussed upstream effects in its September 22, 2009, response to RAI-SRP6.2.2-SPCV-23. For ADS-4 discharge and break locations in the loop compartment, the limiting flowpath is the 2.3 m (7.5 ft) wide corridor between loop compartments, which is large enough to preclude debris blockage. For break locations at the maintenance floor elevation, three open stairwells will preferentially drain the break discharge to the sump. Curbs around openings of the two PXS rooms and the CVS room prevent water from entering these rooms. There are no LOCA break locations inside the CVS room, but such locations are present in the PXS room. If a break occurs in the PXS room, the room will fill and overflow onto the maintenance floor elevation, where the water will drain to the sump through the open stairwells. Breaks in the pressurizer line will either flow to the refueling cavity or to the IRWST, while a break in the passive residual heat removal heat exchanger tube will flow to the IRWST. Initially, water in the refueling cavity will drain to the containment sump, but this gravity-driven drain will cease when the water level in the sump increases enough to close the check valves on this line. An increase to the level of water in the IRWST could overflow into the refueling cavity, which will then drain to the containment sump as previously described.

For the AP1000, the potentially significant choke points for flow holdup are the gravity-driven drain lines and check valves in flowpaths between containment compartments. These drain lines and check valves are those in the refueling cavity drain lines, the PXS-A drain line, the PXS-B drain line, and the CVS compartment drain line. If recirculation water flow is restricted by any of these lines or valves, excessive amounts of water may be held up in the compartments and cavities, and the floodup level in the containment, which affects the gravity-driven core cooling flow, could be adversely affected. The staff asked about the issue in RAI-SRP6.2.2-SPCV-27 and RAI-SRP6.2.2-CIB1-31, and the applicant responded in letters dated January 29, 2010, and June 30, 2010. In these responses, the applicant stated that absent a LOCA in the PXS-A, PXS-B, and CVS rooms, there is no forward flow to transport debris to the lines that would prevent closure of the check valves. The staff verified that these valves are periodically stroke tested to ensure that the valves are capable of adequately reseating. The staff finds this acceptable because no debris would exist in these drain lines. Other sources of leakage into the rooms, such as through cracks in the wall, were determined to

be insignificant. If an LOCA occurs in one of the PXS rooms (there are no break locations in the CVS room), the room would flood and debris could enter the drain lines. However, in this event the valves do not have to function, because the long-term cooling analysis includes cases that model a lower flood level to specifically account for water holdup resulting from a break in the PXS room. Even though the check valves in the drain lines are not expected to leak any significant amount, the applicant's wall-to-wall flooding analysis assumes that each line leaks 26.5 liters per minute (Lpm) (7 gpm). The staff finds that this is a conservative assessment.

For the refueling cavity drain lines, the check valves may be required to reclose after initially flowing forward following an LOCA and the water in the cavity may contain some debris. The applicant stated that the debris would not excessively erode or plug the valve, because the debris is a limited amount of latent and MRI debris and there is time for significant amounts to settle out. The applicant also stated that the debris could include some MRI fine particles that could pass through the downward-turned elbow drain line inlet and that this debris would not likely collect in the check valves. The staff finds that this is acceptable, because very little of the debris in the refueling cavity would pass into the drain lines and valves and would not significantly impede necessary drainage.

To address the issue of reclosure of these check valves after a period of initially flowing forward with debris in the lines, the applicant performed an analysis demonstrating that adequate long-term cooling was available even with a relatively large back leakage through this line. The analysis included an evaluation of Sensitivity Cases 3 and 10 from APP-PXS-GLR-001, which demonstrated that the margin in the original analysis, coupled with the long time required for the containment sump to leak through the check valve into the reactor cavity, was more than sufficient to remove the decay heat. The staff finds that the assumed leakage for this analysis is acceptable because the valves will be periodically stroke-tested to ensure that they acceptably reseat before debris conditions occur, and because the extent to which each closed check valve would have to be held open for this leakage to occur is significantly greater than what would be expected for these debris conditions. The staff reviewed the supporting calculation notes during its June 1 audit of APP-PXS-M3C-012, Revision 1, "Post-LOCA Refueling Canal Drain Check Valve Leak Evaluation," dated May 2010. The staff found the assumptions used to calculate flood levels, decay heat, and reduced core flow to be appropriate and conservative, and the staff agrees that the analysis demonstrates adequate core cooling flow with the assumed leakage.

The only potential choke point for flow to the IRWST is at the IRWST gutter, which extends around the entire containment circumference and drains to the IRWST through two 10-cm (4 in) pipes. Although there is a rough screen on top of the gutter to prevent large debris from entering, even if both discharge pipes become clogged, the only holdup volume will be water inside the gutter. This is because any excess water will spill over to the containment sump and recirculate through the containment recirculation screens. The applicant stated in its May 13, 2009, response to RAI-SRP6.2.2-SPCV-18 that the gutter volume is 0.96 m^3 (34 ft^3), corresponding to an IRWST water level change of 0.5 cm (0.2 in), which the staff finds is insignificant with respect to the recirculated volume.

In conclusion, the staff finds the applicant's evaluation of upstream effects acceptable.

6.2.1.8.2.13 Ex-Vessel Downstream Effects

In Section 6.2 of the AP1000 DCD, the applicant described the potential effects of sump debris on containment and core cooling, including the ex-vessel downstream effects. The term

"downstream effects" refers to the effects of debris that passes through the recirculation screens on systems, structures and components located downstream of the recirculation screens. These effects have been evaluated for operating plants using data and methods developed by the PWR Owners Group. For the AP1000 plant, the applicant performed both an ex-vessel and an in-vessel evaluation for the AP1000 downstream effects. The ex-vessel evaluation describes the effects of debris on the system and components outside the core. This evaluation looks specifically at the disruption of the long-term core cooling flowpath (outside the core) by debris. Section 6.2.1.8.2.9 of this report addresses the staff's evaluation of in-vessel downstream effects (inside the core) for the AP1000.

The staff reviewed several documents, including information provided in the DCD and Westinghouse topical reports, TRs, and other letter reports related to the AP1000 design. Additionally, in an SER dated December 20, 2007, for ANSI/ANS-51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," issued 1983, the NRC evaluated the methodology in WCAP-16406-P, Revision 1, for addressing downstream effects (both in-vessel and ex-vessel) of debris on the performance of containment and core cooling systems in existing U.S. PWRs. TR-26 addresses the applicability of WCAP-16406-P, Revision 1, to potential ex-vessel downstream effects for the AP1000 design.

As stated in TR-26, the data and methods used by the applicant to evaluate ex-vessel downstream effects are outlined in Revision 1 of WCAP-16406-P. The applicant stated that the evaluation methods identified in WCAP-16406-P, Revision 1, that are applicable to ex-vessel long-term core cooling recirculation flowpaths associated with the AP1000 design include valve evaluations for plugging and erosive wear, as described in Sections 7 and 8 and Appendix F of WCAP-16406-P, Revision 1. The screening criteria for valves that are identified in Revision 1 to WCAP-16406-P are applicable to valves in the long-term core cooling recirculation flowpath of PWRs. The applicant stated that only the explosively actuated (squib) valves in the post-LOCA flowpath are not covered by the screening criteria, but that once the squib valves are open, they very closely exhibit the characteristics of a standard gate valve.

The applicant stated that the AP1000 has design features that eliminate the need for downstream effects evaluations of components that are included in Revision 1 of WCAP-16406-P. Those evaluations excluded in the applicant's evaluation of the AP1000 design and the bases for their exclusion are as follows:

- Pump evaluations, including hydraulic performance, disaster bushing performance, and vibration analysis. Because of its passive design features, there are no safety-related pumps in the AP1000 passive core cooling flowpaths to evaluate.

- Heat exchanger evaluations for both plugging and erosive wear. There are no safety-related heat exchangers in the AP1000 passive core cooling flowpaths.

- Orifice evaluations for plugging and erosive wear as described in Sections 7 and 8 and Appendix F of WCAP-16406, Revision 1. There are no orifices in the post-LOCA recirculation flowpath of the AP1000 design.

- Settling of debris in instrumentation lines as described in Section 8 of WCAP-16406, Revision 1. There are no instrumentation lines used in the AP1000 post-LOCA containment recirculation flowpath design that are required to support a safety-related function; they are therefore excluded from consideration.

- Containment spray system. The AP1000 does not have a conventional containment spray system. The nonsafety containment spray function is not permitted to be used during a DBA and is therefore excluded from consideration for the AP1000 design.

As documented in TR-26, use of the applicable methods and models in WCAP-16406-P, Revision 1, consistent with the applicable amendments, limits, and conditions of the associated NRC safety evaluation on WCAP-16406-P, demonstrates that the AP1000 PXS equipment used in post-LOCA recirculation is acceptable for the expected debris loading in the recirculating fluid resulting from a postulated LOCA.

The staff finds that the applicant's evaluation of the ex-vessel downstream effects for the AP1000 design addressed the piping and valves in the recirculation path of the PXS. The methodology and acceptance criteria used are described in WCAP-16406-P, Revision 1, and are consistent with the applicable amendments, limits, and conditions of the NRC safety evaluation for WCAP-16406-P, Revision 1, and are, therefore, acceptable to the staff.

The applicant identified equipment in the post-LOCA flowpath using current piping and instrumentation diagrams (P&IDs) for the AP1000 PXS. The AP1000 PXS P&IDs show no pumps, heat exchangers, orifices, and spray nozzles in the PXS. Therefore, although included in the method of WCAP-16406-P, Revision 1, the applicant's evaluation of the AP1000 PXS does not address pumps, heat exchangers, orifices, spray nozzles, and instrumentation tubing because these components and features are not included in the design of the AP1000 PXS. The applicant listed and described components that are in the AP1000 long-term core cooling flowpath, including both the containment recirculation flowpath and the IRWST injection flowpath.

The downstream path evaluation includes the effects of erosion and wear of the component materials and plugging of the various components. In order to apply erosive and abrasive wear rate models, the debris size and concentration were first assessed. The applicant evaluated the debris types and stated that the debris was composed of latent fibrous and particulate material, with a small amount of coatings debris. In RAI-SRP6.2.2-CIB1-07, the staff raised questions regarding: (1) the results of the applicant's evaluation of the piping system for plugging and wear; (2) the composition of possible debris other than the evaluated latent and coatings debris; and (3) the effects the composition may have on the downstream flowpath. In a response dated November 11, 2008, the applicant clarified that there is no other debris that would be transported to the recirculation screens. The staff finds that the applicant's assumed composition of latent and coating debris for the ex-vessel downstream evaluation is consistent with the debris generated from coatings and the chemical effects evaluated by the staff for the AP1000.

Each identified valve in the PXS was evaluated for plugging and wear against the applicable initial screening criteria in WCAP-16406-P, Revision 1. The PXS consists of open gate, check, and squib valves, all of which are greater than 2.54 cm (1 in) based on their individual flow line diameters. Therefore, according to the initial screening criteria, the valves do not need further evaluation for plugging or wear. However, the criterion to be greater than 2.54 cm (1 in) is based on the assumption that only debris of a certain size will pass through the containment recirculation screen because of the size of its holes. For a limiting direct-vessel-injection-line-break LOCA in the AP1000 design, the water level in the containment permits direct entry of debris-laden water into the DVI line at the break location. This could result in a significantly higher concentration of debris and larger pieces of debris

entering into the core cooling flowpath than if all cooling water first passed through the screen. Therefore, in RAI-SRP6.2.2-CIB1-08, the staff asked about the assumed concentration and composition of debris for the DVI-line-break LOCA.

In a response dated November 6, 2008, the applicant stated that the only ex-vessel components that would see unfiltered containment water during a DVI line break are the 17.3 cm (6.8 in) ID DVI pipe and the reactor vessel DVI nozzle that has a 10.16 cm (4-in) ID venturi. The applicant stated that these components have openings that are large enough that plugging will not occur and that wear of the venturi is not an issue because the purpose of the venturi is only to limit high-pressure blowdown flow, not to restrict recirculation. The staff finds that the applicant has provided reasonable assurance that plugging and wear of components because of debris-laden water in the ex-vessel PXS flowpaths are not significant concerns for the AP1000 design.

All instrumentation sensors in the PXS recirculation lines are strapped to the outside of the piping. Therefore, there are no instrumentation tubes or sensing lines to evaluate for potential debris collection in the tubes or sensing lines. In RAI-SRP6.2.2-CIB1-09, the staff asked about the possible effects of debris, chemicals, and gases in the recirculation water on the accuracy of these strapped instruments as a result of changing the velocity of sound in the fluid. In a response dated November 6, 2008, the applicant clarified that the only strapped-on instruments are temperature sensors that are not affected by the velocity of sound in the fluid. The staff finds that the applicant's clarification adequately addressed the staff's question.

The applicant also evaluated the potential debris collection in the PXS flowpath piping. Based on the minimum flow rates for the PXS flow lines, the applicant determined that the transverse velocity is sufficient to prevent debris settlement in the PXS flow lines; therefore, blockage in PXS flow lines from the settling out of debris would be precluded. In RAI-SRP6.2.2-CIB1-10, the staff asked about flow rates that could be less than the minimum value assumed (e.g., during system flow initiation or realignment) and whether significant debris settlement could occur that would prevent necessary system core cooling flow. In a response dated November 6, 2008, the applicant responded that with the small amount and low concentration of debris that would be present following a LOCA and the large-diameter piping, any settling out of debris, even during very small flow, such as during startup or realignment, would have a negligible effect on PXS flow resistance. The staff finds this acceptable and agrees that because of the very low concentration of debris in the water, only a very small amount of debris could settle out during low flow conditions and would not cause significant blockage.

The applicant credited only passive systems for core and containment cooling; however, the AP1000 design includes a nonsafety active system (the RNS) that could be used for removing core and containment heat during various plant conditions, including a LOCA. The staff evaluated the possible effects of operation of these nonsafety active system ex-vessel downstream components and their capability to remove heat for long-term core cooling. As a result, in RAI-SRP6.2.2-CIB1-01, the staff asked about the use of these systems and: (1) the effects of possible additional amounts of debris ingested as a result of use of the active systems; (2) how ingested debris could affect the capability of these active systems when relied on for long-term cooling; (3) how ingested debris could affect these active systems for long-term cooling; (4) how ingested debris could affect the pressure integrity, leakage, and containment isolation function of these active systems; and (5) whether leakage through pump seals or other components could increase local dose rates so that credited operator actions, if any, would not be met. In a response dated November 11, 2008, the applicant stated that the evaluation of downstream effects has already included the conservative assumption that all latent containment debris could be ingested.

The applicant also stated that it evaluated the RNS ex-vessel downstream components, including pump seals, for wear, abrasion, and erosion using the evaluation methods of WCAP-16406-P for the assumed debris and found the components to perform their functions for a 30-day period beginning at the time of recirculation from the sump. The applicant further stated that in the event of a large source term (like the design-basis core melt source term), the RNS is automatically (i.e., not manually) isolated from the containment. Additionally, the applicant submitted a response dated May 13, 2009, to the staff's followup RAI-SRP6.2.2-CIB1-23 regarding the capability of the RNS isolation valves to close and not leak excessively under debris-laden conditions after the RNS has been functioning. The applicant performed an evaluation of the effects of wear, abrasion, debris loading, and erosion and concluded that the isolation valves will close and not leak excessively under these conditions. The staff finds that this adequately addresses the concerns regarding the evaluation of ex-vessel downstream RNS components under debris-laden conditions.

In RAI-SRP6.2.2-CIB1-11 and RAI-SRP6.2.2-CIB1-12, the staff asked about possible blockage of the ex-vessel downstream flowpath into the vessel and out of the vessel back to the break location as a result of settling or precipitation of boric acid and other chemicals. In responses dated November 6 and 11, 2009, the applicant stated that the concentration of boron and other chemicals is low enough that their precipitation would not occur over the 30-day mission time. The staff finds that this adequately addresses the issue of potential blockage of ex-vessel core cooling water flowpaths by boron or chemical precipitation.

The applicant performed an evaluation of the effects of the possible collection of noncondensable gases in high points in the PXS flowpath. Gases in sufficient quantities that collect and are trapped at high points could cause unacceptable pressure losses and restriction of system cooling flow, especially in a gravity-driven system. In RAI-SRP6.2.2-CIB1-13, the staff asked about the possible collection of noncondensable gases in the PXS flowpath that could impede cooling flow. In a response dated November 11, 2008, the applicant stated that the CMT and passive residual heat removal heat exchanger circuits are not susceptible to gas accumulation during preaccident standby conditions, except in an engineered high point pipe stub that has redundant level sensors. A low concentration of hydrogen is dissolved in the RCS, which is separated from these circuits by a single valve, but very little pressure is required to maintain the hydrogen in solution. The accumulator circuits are also expected to contain nitrogen dissolved in the water, but there is substantial dP following a LOCA such that any gas pockets would not impede accumulator flow. Once the accumulator water is emptied, nitrogen from the tank will be injected into the RCS. This nitrogen is readily vented from the RCS through the open ADS valves, and integral testing performed for the AP1000 showed that this nitrogen injection did not adversely affect the plant and system performance. The recirculation flowpaths from the IRWST to the RCS or from the recirculation screens to the RCS either contain water all the time or, if they contain air, they are short, straight horizontal pipes, such that the air is expelled and will readily fill with water. The staff finds this acceptable for addressing the possible effects of noncondensable gases, other tha
n those gases that could form as a result of chemical reactions or gases that come out of solution at higher accident-condition temperatures, which are discussed below.

Noncondensable gases that could form in the ex-vessel recirculation flowpath as a result of chemical reactions and additional gases that may come out of solution at higher accident temperatures were also evaluated. In RAI-SRP6.2.2-CIB1-23, the staff asked about the possible effects such gases could have in restricting cooling flow. In a response dated November 11, 2008, the applicant stated that the amount of gas as a result of chemical

reactions is small and is limited by the small amount of materials that could react with the coolant. The amount of gases that could form by chemical reactions and that could come out of solution at higher temperatures would have time to bubble to the surface in the pool and be released to the containment. The staff finds this acceptable, because the amount of these gases is limited in quantity or they will have time to come out of solution in the pool water before recirculating in the core cooling flowpath.

Another issue of concern to the staff is how the squib valves may differ from the evaluated gate valves and the effects that squib valve propellant residue or chemicals could have on the ex-vessel downstream flowpath. The staff notes that the actuation of the squib valves occurs when there is no debris in the valve with which these chemicals could interact, such that a combined effect of both chemicals together with debris is not possible. However, the effects of the residue or chemicals as they mix with the system fluid without debris could also be an issue. Therefore, in RAI-SRP6.2.2-CIB1-25, the staff asked about the differences from the gate valve design and the possible effects the residue or chemicals could have in impeding the recirculation flow through the valves. In a response dated September 22, 2009, the applicant stated that both the squib valves and gates valves have sufficiently large flow openings and internal crevices, such that latent and particle debris would not get caught or restrict the flow. Regarding the effects of the residue and chemicals, the applicant stated that each of the 12 squib valves in the plant contains less than approximately 300 g (0.66 lbs) of propellant in which potassium is the predominate constituent. If all of the resulting potassium were mixed with the minimum post-LOCA sump volume, the concentration of potassium would be approximately 0.5 ppm. The applicant stated that this is much less than the concentration of sodium that would be present, which is chemically similar to potassium. The applicant stated that at this concentration, the potassium would stay in solution, and there would be negligible impact on downstream components. The applicant also stated that the remaining constituents of the propellant include gases in concentrations much lower than the potassium concentration, such that they would have no impact on the downstream components.

In reviewing Tier 1 to the AP1000 DCD, the NRC staff found that Table 2.1.2-4, "Inspections, Tests, Analyses, and Acceptance Criteria," included ITAAC to verify the design and qualification of the as-installed squib valves in the ADS of the AP1000 reactor. However, AP1000 DCD Tier 1, Table 2.2.3-4, "Inspections, Tests, Analyses, and Acceptance Criteria," did not include ITAAC to verify the design and qualification of the as-built squib valves in the AP1000 PXS. In RAI-SRP6.2.2-CIB1-29, the NRC staff noted that the squib valve design and applications in the AP1000 reactor will not be adequately similar to be represented by the tests or type tests for the ADS squib valves. Therefore, the NRC staff requested that the applicant provide additional ITAAC to verify the capability of the different squib valve designs and sizes and their applications in the AP1000 reactor. In its RAI response dated March 12, 2010, the applicant agreed that ITAAC related to the active safety-related valve functions of the PXS containment recirculation squib valves and IRWST injection squib valves should have been included in AP1000 DCD Tier 1, Table 2.2.3-4. The applicant provided a planned revision to AP1000 DCD Tier 1, Section 2.2.3, "Passive Core Cooling System," to address the PXS squib valves. The applicant also provided planned ITAAC for the PXS squib valves to be included in AP1000 DCD Tier 1, Table 2.2.3-4, specifying that tests or type tests will be performed that demonstrate the capability of the valve to operate under its design condition, and that an inspection will be performed for the existence of a report verifying that the as-installed squib valves are bounded by the tests or type tests. Revision 18 to the AP1000 DCD included the modifications described in the RAI response. The NRC staff finds that AP1000 DCD Tier 1, Section 2.2.3, and Table 2.2.3-4, provide acceptable ITAAC to verify the design and qualification of the as-built PXS squib valves. Therefore, RAI-SRP6.2.2-CIB1-29 is resolved.

The applicant's evaluation of the effects of the residue and chemicals on the sump pool water is acceptable to the staff, since the concentrations are very small. However, the residue and chemicals from the valve actuation initially would be introduced into a much smaller volume of water inside the valve and in the immediate vicinity of the downstream pipe. To resolve this issue, the staff is reviewing the specific qualification testing of the squib valves being performed by the applicant. The staff has observed actual testing of prototype squib valve designs and will also observe testing of the final squib valve designs before their installation in the plant. Based on the staff's observations, the amount of residue and chemicals entering the flowstream area is small and would not likely restrict the cooling flow through the squib valves. Therefore, the staff finds the applicant's evaluation and the squib valve qualification testing program to adequately address the possible effects of squib valve residue and chemicals on the ex-vessel downstream flowpath.

In conclusion, the staff finds the applicant's evaluation of ex-vessel downstream effects acceptable.

6.2.1.8.3 Conclusion

In summary, the staff reviewed the proposed DCD changes and associated COL requirements to establish the adequacy of the IRWST and containment recirculation sump screen performance. Based on the evaluation described in the foregoing sections, the staff concludes that the design and analyses satisfy the requirements in GDC 35, GDC 38 and 10 CFR 50.46(b)(5).

6.2.2 Passive Containment Cooling System

6.2.2.1 Summary of Technical Information

Revision 17 of the DCD makes several changes to the text and figures in Section 6.2.2, and Change Number 70, included in a letter dated July 6, 2010, proposes further changes to the figures. As a result of these changes, the applicant also revised DCD Tier 1, Figure 2.2.2-1, "Passive Containment Cooling System Piping and Instrumentation Diagram (PCS P&ID)," and Table 2.2.2-2, as described and evaluated below.

6.2.2.2 Evaluation

Changes made to Figure 6.2.2-1 are evaluated as follows:

- Two passive containment cooling water storage tank (PCCWST) discharge lines were combined to a single larger line. TR-103, "Fluid System Changes," issued May 2006, stated that this change was necessary in order to achieve the flow rates required for adequate containment cooling and was not a functional change. In a July 18, 2008, response to RAI-SRP6.2.2-SPCV-01, the applicant confirmed that the flow rates reported in Table 6.2.2-1 were unaffected; therefore, the staff finds this piping change to be acceptable.

- The two PCCWST narrow-range pressure-based level sensors that shared taps with the wide-range level sensors were replaced with two inside wall-mounted ultrasonic, noncontact level sensors. TR-103 reported that this change was made in order to enhance accuracy over the wide-range level measurement at the top of the tank. Based

on DCD Tier 1, Table 2.2.2-1, the narrow-range sensors are not safety related; therefore, the change does not impact the safety design basis for the PCS. Additionally, the change to ultrasonic sensors provides diversity in measurement of the PCCWST level because the nonsafety-related ultrasonic sensors now differ from the safety-related dP sensors. The staff finds this change acceptable.

- A second makeup line to the PCS water distribution bucket was added to provide the piping separation required to support a beyond-DBA scenario. Because a safety-related line already exists to supply the water distribution bucket, this new line is nonsafety related. The staff finds this change acceptable because it has no impact on the safety design basis for the PCS.

- As described in TR-103, a nonsafety-related spray system was added to the spent fuel pool (SFP) in response to a National Academy of Sciences study on the potential danger if water were drained from the SFP. The line to the new spray system is branched from the existing PCS makeup line to the SFP, and the location where this existing line stems from the PCCWST is changed from a 15.24 cm (6 in) standpipe to the bottom of the tank. A normally closed manual isolation valve was added to provide a boundary between the ASME Code, Section III Code Class 3 SFP makeup line and the nonsafety-related SFP spray header. Both this valve and the existing normally closed safety-related manual isolation valve must be opened in order to activate the SFP spray. In Revision 0 of its response to RAI-SRP6.2.2-SPCV-02, dated July 18, 2008, the applicant stated that the use of the SFP spray was controlled by procedure and not allowed during a DBA; therefore, there would be no impact on PCS performance. The staff requested additional information to determine if PCCWST inventory could be distributed to the SFP when it was needed for containment cooling. In Revision 1 of its response to RAI-SRP6.2.1.1-SPCV-06, dated August 31, 2009, the applicant added a paragraph to DCD Section 6.2.2.4 stating that the use of the PCCWST to provide water to the SPF spray header would be governed by the Extensive Damage Mitigation Guidelines included in NEI 06-12, "B.5.b Phase 2 & 3 Submittal Guideline," issued December 2006, and this document was added as Reference 33 to DCD Section 6.2.7. The staff finds this acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- As described in a letter dated July 6, 2010, Change Number 70 revises the P&ID to accommodate lower than expected efficiency of the PCS recirculation pumps. These changes, which include increasing the size of the recirculation pump piping and associated valves, and adding a line to bypass the pumps, ensure that the ancillary diesel generator does not exceed the steady-state load limit. These changes are acceptable because they have no impact on the system or component design evaluated in NUREG-1793, Section 6.2.2.

- Several changes were made to the P&ID that the staff agrees have no impact on the evaluations made in NUREG-1793, Section 6.2.2. These include moving the recirculation heater from downstream to upstream of the chemical addition tank, standardizing flow orifice flange sets to 1.9 -cm (0.75–in) piping components, and correcting errors in the certified P&ID.

The changes made in the Revision 17 of the DCD to Tier 2, Figure 6.2.2-2, "Simplified Sketch of Passive Containment Cooling System," and Tier 1, Figure 2.2.2-1, are consistent with the items described above. The staff finds them acceptable.

DCD Tier 1, Table 2.2.2-1, identifies the components that perform safety-related functions for the PCS, and Table 2.2.2-2 identifies the PCS safety-related lines. Revision 17 of the DCD added several valves to Table 2.2.2-1 to reflect the P&ID changes previously discussed, but the lines in Table 2.2.2-2 were unaltered. However, the applicant subsequently proposed changes to the lines in this table in its July 18, 2008, and May 27, 2009, responses to RAI-SRP6.2.2-SPCV-04 and RAI-SRP6.2.2-SPCV-20. The changes made to both Tier 1 tables were found to be acceptable because they are consistent with changes made to Tier 2, Figure 6.2.2-1, and with each other. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In the certified DCD, Section 6.2.2.4 states that the frequency of operational testing of the PCS was consistent with the plant TS and the inservice testing program. Revision 17 of the DCD removes the reference to the plant TS. However, the applicant's May 27, 2009, response to RAI-SRP6.2.2-SPCV-03, Revision 1, withdraws this change. The staff finds this acceptable because the TS are used to define some of the operational test frequencies. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

6.2.2.3 Conclusion

The staff finds the proposed changes to the DCD acceptable because they have no impact on the statements or conclusions of NUREG-1793, Sections 6.2.1.6 and 6.2.2, regarding primary containment testing and inspection and the PCS, respectively.

6.2.3 Shield Building Functional Design

6.2.3.1 Summary of Technical Information

Modifications were made to the shield building to strengthen it against additional external hazards, to make it more robust to seismic events, and to simplify construction. These changes, which include alterations to the inlet and outlet of the air flowpath and the height of the shield building, are expected to impact the air flow rate through the PCS. APP-GW-GLR-096, submitted August 10, 2010 includes a description of these changes and the impact they have on existing test reports. It also documents how the changes were incorporated into a WGOTHIC model that was used to run the limiting DBA and beyond design basis accidents (BDBA) affected by these changes. Appendix A of APP-GW-GLR-096 includes proposed DCD changes.

6.2.3.2 Evaluation

The staff evaluation of APP-GW-GLR-096 will be documented in Chapter 23.

6.2.3.3 Conclusions

The impact of the shield building changes on the containment functional design capability and the staff's conclusion will be provided in Chapter 23.

6.2.4 Containment Isolation System

The major function of the containment isolation system (CIS) is to isolate the containment to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary. The CIS consists of the piping, valves, and actuators that isolate the containment.

In a letter dated April 5, 2006, the applicant submitted a request to modify the AP1000 DCD Tier 2 information to incorporate the ability to inject a small quantity of zinc acetate into the RCS. TR-32 describes this proposed change.

Zinc acetate would be added using the same piping and valving as the CVS hydrogen addition piping, which contains two containment isolation valves, CVS-PL-V092 and CVS-PL-V094. This would also require changing the normal position of the outside containment valve, CVS-PL-V092, from normally closed to normally open. The inside containment isolation valve, CVS-PL-V094, is a normally closed check valve. The proposed hardware change would also replace a portion of the 2.54 cm (1 in) pipe (downstream from the inside containment isolation valve) with a heavier wall 1.27 cm (0.5 in) pipe. The staff has reviewed and approved this change (Section 5.2.3 of this report).

6.2.4.1 Summary of Technical Information

The modification described in TR-32 results in changing the outside containment isolation valve, CVS-PL-V092, from normally closed, failed closed to normally open, failed closed.

Revision 17 of the DCD provides four additional overpressure relief valves between two normally closed containment isolation valves, identified in Table 6.2.3-1, "Containment Mechanical Penetrations and Isolation Valves." These valves have also been added to Tier 1, Figure 2.2.1-1, and identified in Tier 1, Table 2.2.1-1, as CCS-PL-V220, SFS-PL-V067, VWS-PL-V080, and WLS-PL-V058.

6.2.4.2 Evaluation

The normal position of the containment isolation valve, CVS-PL-V092, in the CIS would change from normally closed to normally open. The valve would still fail closed, maintaining its containment isolation function. This change is indicated in Revision 17 of the DCD, Tier 2, Table 6.2.3-1, and shown in Figure 9.3.6-1, Sheet 1 of 2. The following functions and properties are not affected: the containment isolation function signal, the containment isolation design, the valve designation as active (Table 3.9-12 in the DCD), the safety-related mission, the inservice testing type and frequency requirements (Table 3.9-16), and the valve functional requirements for containment isolation (Tier 1, Table 2.3.2-1).

GDC 55, "Reactor Coolant Pressure Boundary Penetrating Containment"; GDC 56, "Primary Containment Isolation"; and GDC 57, "Closed Systems Isolation Valves," of Appendix A to 10 CFR Part 50 require that containment isolation valves outside containment be located as close to containment as practical. Acceptance Criterion 9 in NUREG-0800 Section 6.2.4, "Containment Isolation System," Revision 3, issued March 2007, also invokes this requirement. In RAI-SRP6.2.4-SPCV-01, the staff asked the applicant to provide the distances from the containment to the outboard isolation valves, and in RAI-SRP-6.2.4-SPCV-02, the staff asked the applicant to add the approved distances to DCD Tier 2, Table 6.2.3-1. The applicant responded to the RAIs with the distances provided in its letter dated January 13, 2009, and with

its commitment in its letter of April 13, 2009, to add these distances to DCD Tier 2, Table 6.2.3-1. The staff found the responses acceptable. The corresponding ITAAC in Table 14.3-7, "Radiological Analysis," that the containment penetration isolation features be configured as given in Table 6.2.3-1, remains acceptable to the staff as written. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In Revision 15 of the DCD, additional requirement M in Section 6.2.3.1.3, "Containment Isolation Design," states the following:

> Containment penetrations with leak-tight barriers, both inboard and outboard, are designed to limit pressure excursions between the barriers due to heating of fluid between the barriers. The penetration will either be fitted with relief or check valves to relieve internal pressure or one of the valves has been designed or oriented to limit pressures to an acceptable value.

Section 6.2.4.2 of NUREG-1793 states, "All overpressure relief valves used as containment isolation valves comply with the SRP acceptance criterion of having a setpoint greater than or equal to 150 percent of the containment design pressure."

The applicant's response to RAI-SRP6.2.4-SPCV-03 confirms that the four new relief valves comply with the NUREG-0800 Section 6.2.4 acceptance criterion of having a setpoint greater than or equal to 610 kPa (88.5 psig), 150 percent of the containment design pressure. These new relief valves are shown in Revision 17 of the DCD, Tier 1, Figure 2.2.1-1, and listed in Tier 2, Table 6.2.3-1.

In RAI-SRP6.2.4-SPCV-03, the staff also asked whether the CVS letdown line at penetration P06 should be similarly provided with an overpressure relief valve between the two normally closed containment isolation valves, CVS-PL-V045 and CVS-PL-V047. The applicant's response of May 20, 2009, indicates that it will add relief valve CVS-PL-V058, which will comply with the design requirements of the relief valves already added. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Among the apparent discrepancies noted in RAI-MISC-SPCV-01 is the fact that the change to the normal position of containment isolation valve CVS-PL-V092 has not been reflected in Section 9.3.6, "Hydrogen Addition Containment Isolation Valve," which should indicate that the valve is normally open, failing closed. Also, the position of CVS-PL-V092 in Table 9.3.1-1, "Safety-Related Air Operated Valves," should read "normally opened, failed closed."

In Revision 16 of the DCD, the applicant also made two editorial changes to Table 6.2.3-1, Sheet 1 of 4. The first identifies containment isolation valves CAS-PL-V015, CCS-PL-V201, CCS-PL-V208, and CCS-PL-V207 and observes the correct naming convention. In addition, for containment isolation valves CCS-PL-V207 and CCS-PL-V208, the applicant corrected the containment isolation signal to "S." The staff agrees with these two editorial changes.

6.2.4.3 Conclusion

The staff finds that the applicant's proposed modification to the AP1000 CVS system, approved in Section 9.3.6, does not adversely affect the containment isolation design and is, therefore, acceptable. The five thermal relief valves provided for overpressure protection are in

accordance with regulatory guidance, consistent with DCD commitments, and are acceptable to the staff.

The editorial changes are acceptable.

6.2.5 Containment Hydrogen Control System

The containment hydrogen control system is provided to limit the hydrogen concentration in the containment so that containment integrity is not endangered.

On September 2004, the staff provided its assessment of the AP1000 hydrogen ignition subsystem design in Section 6.2.5.1 of NUREG-1793. As stated in paragraphs 3 and 9 of Section 6.2.5.1, adequate igniter coverage was provided based on implementation of the igniter location criteria in DCD Table 6.2.4-6.

DCD Tier 2, Table 6.2.4-6, provides the criteria used in the evaluation and the application of the criteria to specific compartments. On the basis of the staff's review and the applicant's implementation of the igniter location criteria as listed in DCD Tier 2, Table 6.2.4-6, the staff concluded that adequate igniter coverage had been provided.

6.2.5.1 Summary of Technical Information

In APP-GW-GLN-003 (TR-37), "Hydrogen Igniter Locations," Revision 1, the applicant modified the elevations or locations of certain hydrogen igniters within the AP1000 hydrogen control system. The applicant stated that the modifications were necessary because either the polar crane elevation or the pressurizer height had been changed, or in order to place the igniters in more easily accessible locations or to avoid trip hazards.

In Revision 16 of the DCD, Figures 6.2.4-5 through 6.2.4-13 show the proposed locations of the hydrogen igniters, and Tables 6.2.4-6 and 6.2.4-7 identify the proposed hydrogen igniter locations. The number of igniters is unchanged at 64.

6.2.5.2 Evaluation

Revision 16 of the DCD, Table 6.2.4-6, provides the criteria used in the evaluation and the application of the criteria to specific compartments. The changes to igniter locations as a result of the continuing COL and detailed design activities for the AP1000 satisfy the igniter location criteria identified in DCD Table 6.2.4-6 (Sheet 1 of 3) that were used for the DC review of the hydrogen igniter subsystem and referenced in the AP1000 SER. Therefore, changes in the placement of the hydrogen igniters that are consistent with the criteria in Table 6.2.4-6 do not alter the design function of the igniters, have no effect on any analysis or analysis method, and do not affect the performance or controls of hydrogen control functions.

On the basis of the staff's review and the applicant's implementation of the igniter location criteria as listed in DCD Tier 2, Table 6.2.4-6, the staff concludes that adequate igniter coverage has been provided.

6.2.5.3 Conclusion

The staff finds that the applicant's proposed modification to the AP1000 hydrogen control system design with respect to the change in hydrogen igniter locations, as described in TR-37, is consistent with the previously approved criteria and, therefore, acceptable.

6.2.6 Containment Leak Rate Test System

The containment leak rate test system is designed to verify that leakage from the containment remains within limits established in the TS.

6.2.6.1 Summary of Technical Information

The containment penetrations, including electrical penetrations, subject to Type B testing appear in Figure 6.2.5-1, "Containment Leak Rate Test System Piping and Instrumentation Diagram." The applicant has added the test connection assembly for the newly added electrical penetration, P03, to the list of electrical penetrations test connections in Figure 6.2.5-1.

6.2.6.2 Evaluation

The design commitment to provide a test assembly for Type B leak rate testing for the newly added electrical penetration, P03, is acceptable.

6.2.6.3 Conclusion

Based on its review, the staff finds the proposed addition of a Type B leak rate test assembly for the new electrical penetration, P03, acceptable.

6.2.8 Tier 1, Chapter 2.2.1, Containment System

6.2.8.1 Summary of Technical Information

In TR-97, APP-GW-GLN-022, Revision 1, "DAS Platform Technology and Remote Indication Change" dated May 2007, the applicant identifies and justifies standard changes to Revision 15 of the DCD. These changes include relocating the diverse actuation system (DAS) squib valve control cabinet (DAS-J3-003) and adding the DAS instrumentation cabinet (DAS-JD-004) to the southern section of the auxiliary building. The DAS is a nonsafety-related system. These changes necessitate the addition of a containment electrical penetration, P03. In a letter dated May 14, 2007, the applicant submitted responses to all the NRC RAIs on TR-97.

6.2.8.2 Evaluation

The staff's assessment of the CIS design was provided in Section 6.2.4 of NUREG-1793. As stated in the NUREG-1793 section, the containment penetration design of isolation barriers met the following acceptance criteria of NUREG-0800 Section 6.2.4.

Containment isolation equipment may be subject to potentially harsh conditions resulting from pressure, temperature, flooding, jet impingement, radiation, missile impact, and seismic response. The staff's review confirmed that the CIS had been properly classified to ensure that protection from these environmental hazards is encompassed by the mechanical and electrical design bases and quality standards of the isolation system.

The CIS will be designed to ASME Code Section III, Class 2 criteria. Containment penetrations are classified as Quality Group B, as defined in RG 1.26, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," Revision 4 and seismic Category I. The containment penetrations are identified as Class B, equivalent to ANS safety Class 2. The staff concluded that the applicant had selected the appropriate mechanical design classification for the CIS.

The staff determined that the CIS met the acceptance criteria of Section 6.2.4 of NUREG-0800, including the relevant requirements of GDC 1, "Quality Standards and Records"; GDC 4; and GDC 16, "Containment Design."

The acceptable design standards for electrical penetrations, namely ASME Code Section III, seismic Category I, non-Class 1E qualified, harsh environment qualified, will also apply to the new electrical penetration, P03, as shown in DCD, Revision 17, Tier 1, Table 2.2.1-1. The addition of the new electrical penetration, P03, is shown in DCD Revision 17, Figure 2.2.1-1 where it is included in the note "1 of 24 and one spare." This is inconsistent with TR-97 and in RAI-SRP6.2.4-SPCV-04, the staff requested the note be corrected. In a letter dated May 17, 2010, the applicant agreed to revise the note to state "1 of 25." In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Tier 1, Table 2.2.3-4, Inspections, Tests, Analyses, and Acceptance Criteria, describes the equipment listed in Tier 1, Table 2.2.3-6, including all electrical penetrations, as having sufficient thermal lag to withstand the effects of hydrogen burns associated with severe accidents. The newly added electrical penetration, P03, has been added to the list of electrical penetrations in Tier 1, Table 2.2.3-6. The design commitment to provide the thermal lag to the newly added electrical penetration, P03, is acceptable.

6.2.8.3 Conclusion

Based on its review, the staff finds the proposed change as described in TR-97, Revision 1, which adds an additional containment electrical penetration in accordance with previously acceptable design criteria for electrical penetrations, namely ASME Code Section III, seismic Category I, non-Class 1E qualified, harsh environment qualified, will also apply to the new electrical penetration, P03. The staff finds that the applicant's proposed modification to the AP1000 CIS design with respect to the addition of an electrical penetration, as described in TR-97, is consistent with the previously approved criteria and is, therefore, acceptable.

6.4 Control Room Habitability Systems

6.4.1 Summary of Technical Information

Section 6.4 of the AP1000 DCD has undergone significant revision. The revisions include a major redesign of the AP1000 main control room (MCR) emergency habitability system (VES). The VES is a passive system design that consists of safety-related canisters of air that supply the control room with fresh, uncontaminated breathing air. The system does not require alternating current (ac) power to function and is required to function for 72 hours. After 72 hours, nonsafety systems can be credited for control room habitability. In the certified design, the COL applicant was responsible for the testing frequency associated with the control room integrity program. Additionally, the certified design provided no filters to remove

radioactivity from the control room environment. The system only replaced the contaminated control room air with bottled air that was uncontaminated. In the amendment, the applicant removed the COL information item and provided a general description of the control room integrity program with the testing frequency. Additionally, to permit the use of the AP1000 design at more sites, the radiation dispersion factors were also expanded. In developing a control room integrity program, which includes in-leakage testing for the control room envelope (CRE), the applicant was unable to establish achievable acceptance criteria for the in-leakage testing with the new radiation dispersion factors. As a result, the applicant made a series of significant design changes to add margin to the control room dose calculations.

The changes included reducing control room in-leakage through various design provisions, including precluding ductwork from penetrating the CRE and reconfiguring the vestibule. A filter train was added to the passive system as well. The filter train consists of an eductor with ductwork, silencers, a particulate filter, and a high-efficiency particulate air (HEPA) filter. The design change resulted in changes to Tier 1, Tier 2, TS, and ITAAC. Additionally, the applicant has made a number of other unrelated changes, including the redesignation of the technical support area to the control support area. This allows COL applicants more flexibility in designating a technical support area. The design of this system has evolved over the course of the review, and numerous applicant submittals describe the changes. The applicant consolidated all the changes into a response to RAI-SRP6.4-SPCV-15 Revision 1, which was submitted in a letter dated May 24, 2010.

The changes can be grouped into six broad categories. The first includes changes associated with addressing the control room integrity program in the DCD. The approved version of the DCD made this the responsibility of the COL applicant and documented it as a COL information item. The second involves changes associated with the new passive filter train. Third are changes associated with the design changes to reduce the unfiltered in-leakage. The second and third categories of changes were necessary because the applicant revised the radiation dispersion factors to expand the scope of sites that would be covered by the certification. The higher dispersion factors required less leakage and more effective control room fission product removal. The fourth category includes changes associated with the redesignation of the technical support center (TSC) as the control support area. The fifth involves changes intended to improve operational flexibility of the system by including four isolable banks of compressed air canister banks rather than a single bank. Last, there are editorial changes.

A number of changes are associated with the removal of the COL information item on the control room integrity program. The applicant provided a description of the control room integrity program in the final safety analysis report (FSAR) using tracer gas testing. The applicant also included TS implementing the program.

A number of changes are associated with the introduction of a passive filter train. An innovative single passive filter train was added to the existing compressed air system. An eductor was designed to connect to the existing compressed air system. The eductor draws in unfiltered control room air and circulates it through a filter train. The filters are designed and tested to meet the intent of RG 1.52, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Post-Accident Engineered-Safety-Feature Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Revision 3, issued June 2001, and design provisions were made to demonstrate that the single passive train meets the single failure criteria.

A number of design changes are intended to reduce unfiltered in-leakage. The control room air exhaust was moved to vent through the vestibule. This created a sort of purge, reducing unfiltered air ingress from the MCR doors. Additionally, the applicant made design commitments to preclude ductwork from penetrating the CRE. Other material changes were made as well.

Changes are associated with the redesignation of the TSC as the control support area. The applicant changed the description in DCD Tier 2, Section 6.4, and the Tier 2 description in Chapter 18.

The applicant made changes to improve operational flexibility. For example, it divided the existing air tanks into four isolable headers to allow maintenance work to be done on some of the tanks while the remainder of the system remains functional. TS were added to address the situation where one bank of compressed air tanks is out of service. The applicant also performed the dose analysis for the fuel handling accident with a shorter time to reactor shutdown. This permits fuel movement earlier than was previously analyzed.

The applicant also made a number of editorial changes. For example, it changed the general information for onsite chemicals identified in Table 6.4-1. The list of chemicals is general and does not provide the quantity of material or the distance to the control room intake. Additionally, the COL applicant is responsible for identifying and evaluating the onsite and offsite chemicals. As a result, the staff considers this change to be editorial. This SER does not describe these types of changes in detail. With the exception of the editorial changes, a more detailed evaluation of the major changes is provided below.

6.4.2 Evaluation

6.4.2.1 Evaluation of Control Room In-Leakage Testing

In developing a control room integrity program, the applicant had difficulty establishing an achievable in-leakage that could be demonstrated through testing. Originally, the applicant proposed a total effective in-leakage of 0.0425 cubic meters per minute (m^3/min) (1.5 cubic feet per minute (cfm)). This would account for both unfiltered in-leakage as well as effective in-leakage though the doors. The staff issued a number of RAIs to determine how this design limit would be demonstrated. In response, the applicant made a major redesign of the VES. In a letter dated May 24, 2010, the applicant responded to RAI-SRP6.4-SPCV-15 R1 and included all the DCD changes associated with the VES redesign. The applicant has removed the analysis assumption of 1.5 cfm effective unfiltered in-leakage from the MCR dose analysis. The analysis assumption was revised to assume 5 cfm unfiltered in-leakage into the control room as a result of ingress/egress activities. An unfiltered in-leakage of 5 cfm is appropriate for the AP1000 control room because of the incorporation of a two-door vestibule. The control room doses with this increased effective in-leakage assumption required the addition of a passive filtration line to the VES to remain below regulatory limits.

The NRC issued GL 2003-1, "Control Room Habitability," dated June 12, 2003, to alert addressees to findings that the control room licensing and design bases and applicable regulatory requirements may not be met, and that existing specification surveillance requirements may not be adequate. In 2006, the staff approved a modification to the Standard Technical Specifications (STS) (NUREG-1430, "Standard Technical Specifications-Babcock and Wilcox Plants"; NUREG-1431, "Standard Technical Specifications-Westinghouse Plants"; NUREG-1432, "Standard Technical Specifications-Combustion Engineering Plants";

NUREG-1433, "Standard Technical Specifications-General Electric Plants (BWR/4)"; and NUREG–1434, "Standard Technical Specifications-General Electric Plants (BWR/6)") that were proposed by the PWR and boiling-water reactor (BWR) owners groups' Technical Specification Task Force (TSTF) in STS change traveler TSTF-448, Revision 3. The notice of availability for adopting TSTF-448, Revision 3, using the consolidated line item improvement process was published in the *Federal Register* on January 17, 2007 (72 FR 2022). TSTF-448, Revision 3, addresses the tracer gas surveillance, adding a TS action for an inoperable CRE and instituting a CRE habitability program that will ensure that CRE habitability is maintained.

In Revision 17 of DCD Tier 2, Sections 6.4.5.1 and 6.4.5.4 commit to performing tracer gas testing during preoperational inspection and testing, and periodically during the life of the unit in accordance with RG 1.197, "Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors," issued May 2003; they also commit to performing followup self-assessments. However, the AP1000 generic TS did not include a surveillance requirement to measure unfiltered in-leakage into the CRE (the tracer gas test), and required actions for an inoperable CRE boundary and a CRE habitability program as approved by the NRC in STS generic change TSTF-448, Revision 3.

In RAI-SRP6.4-SPCV-01, the staff asked the applicant to incorporate the changes to the STS made by TSTF-448 in AP1000 DCD Tier 2, Chapter 16, "Technical Specifications." In a letter dated May 4, 2009, the applicant responded to incorporate the DCD changes according to the staff's request.

Additionally, the staff raised questions with the applicant about a demonstrable control room in-leakage design basis. VES is a passive system design. There is no safety-related emergency electric diesel to provide electrical power during DBAs accompanied by a loss of outside power. Therefore, there is not enough Class 1E power to drive fans to recirculate control room ventilation air through filters to remove activity during radiological accidents. There is a limited supply of bottled compressed air to maintain control room habitability during DBAs. Based on DCD Revision 17, the VES design had no filtration trains to recirculate and filter the air in the CRE. The bottled air can only supply 1.84 m^3/min (65 cfm) for 72 hours. To meet the dose limits, the unit could only accept 0.0425 m^3/min (1.5 cfm) MCR in-leakage (MCR unfiltered in-leakage and MCR doors ingress/egress combined) to maintain operator dose rates below the required levels. Safety class calculation "AP1000-LOCA Dose Analysis" is the key analysis to demonstrate that MCR operator dose is below 5 roentgen equivalent man (rem) total effective dose equivalent (TEDE), as dictated by GDC 19, "Control Room," in Appendix A to 10 CFR Part 50. Should the MCR in-leakage exceed the 0.0425 m^3/min (1.5 cfm) design limit by a small amount, the 5 rem TEDE limit would be exceeded. The safety margin for this configuration was very small.

The applicant chose to ease the safety margin issue for the VES passive system design by installing a passive filtration line to the VES system. The bottled air will provide 1.84 m^3/min (65 cfm) to an eductor to induce at least 17.0 m^3/min (600 cfm) of air from the MCR into a filter bank to remove radionuclides. This will allow 0.425 (15 cfm) MCR in-leakage and keep MCR operator dose below the required 5 rem TEDE. By doing this, the safety margin issue of the VES passive system design is relaxed. The resulting system is a single safety Class 3, passive filtration line, which is new to the ASME Code. Using the passive filtration line, the VES is able to maintain operator dose below the GDC 19 requirements assuming a total in-leakage of 0.425 m^3/min (15 cfm). The dose analysis assumes that 5 cfm results from ingress/egress activities and up to 0.283 m^3/min (10 cfm) of in-leakage can occur from all other sources. The 0.283 m^3/min (10 cfm) in-leakage is verified through the control room integrity leakage program.

As a result, the applicant has established an achievable design basis in-leakage, and it has properly accounted for effective in-leakage through the doors. Additionally, there is a control room integrity program that meets the recommendations in RG 1.197, and there are TS consistent with TSTF-448. Therefore, the staff finds the applicant's approach acceptable.

6.4.2.2 Evaluation of the Passive Filter Train

The VES uses a bank of compressed air storage tanks to provide the MCR with breathable air and maintain a positive pressure relative to its adjacent areas during accident conditions. The system is designed to deliver a constant flow of 1.84 ± 0.14 standard m^3/min (65 ± 5 standard cubic feet per minute (scfm)) for 72 hours. Using the current VES design, the applicant developed a passive air filtration line that uses an eductor to induce a filtration flow through the MCR of at least 17.0 standard m^3/min (600 scfm). The components that comprise the passive filtration portion of the VES are located entirely within the MCR envelope.

To the extent applicable, the filtration line is designed in accordance with RG 1.52, Revision 3. The applicant added a conformance assessment to Appendix 1A to the DCD to compare the passive filtration line to the requirements defined in RG 1.52, Revision 3, and added this RG to DCD Table 1.9-1. In the conformance assessment, the applicant took an exception to Regulatory Position C.6.1. The staff asked the applicant to explain why it needed an exception to this regulatory position. In a letter dated May 24, 2010, the applicant provided a revision to its response to RAI-SRP6.4-SPCV-15 that stated that the filtration line conforms to Regulatory Position C.6.1. It included the associated changes to Appendix 1A to the DCD to reflect this conformance. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The filtration line comprises an intake grill located near the inner vestibule door inside the MCR envelope. This location was chosen because it is expected to be the location where the greatest amount of in-leakage occurs as a result of ingress/egress activities. Flow is then directed through safety-related ductwork and a silencer into the eductor. The VES supply line is connected to the eductor. The VES bottle air supply is the motive force that drives at least 600 cfm of air flow through the intake duct. The eductor works by directing a small amount of compressed air through a nozzle along the walls of the opening. When released into the nozzle, this small amount of compressed air is moving at near sonic speeds and creates a powerful vacuum in the area upstream of the nozzle. This vacuum draws air in through the surrounding duct and pulls it through the nozzle. The compressed air then carries it downstream away from the eductor. The eductor then directs the flow through a second silencer, a HEPA filter, a charcoal filter, and a postfilter. The filtration units work to remove particulates and iodine from the air to reduce the potential MCR dose. Redundant flow instruments are located downstream of the filtration units to ensure that adequate flow is passing through the filtration units during testing activities. The new flow instrumentation has high and low alarms to alert the operator of possible filtration issues during testing. After passing the flow instrumentation, the filtered air is discharged to three locations inside the MCR envelope. Approximately 1.70 to 1.98 m^3/min (60–70 cfm) would be discharged in the vicinity of the shift supervisor's officer or operator break room, and the remaining 17.0 m^3/min (600 cfm) would be discharged into the main control area through two discharge paths on the opposite side of the control room from the air intake located in the operations work area. Two flow dampers located downstream of the postfilter control the flow distribution.

Although each of the components in the filter train arrangement has been used before, this particular application is novel in the nuclear industry. The applicant constructed a scale model to demonstrate that the system would function. The scale model was tested, and the test results were presented to the NRC in a public meeting on December 15, 2009. The tests demonstrated that the system would function as designed.

The existing pressure-regulating valves in the VES control pressure and flow from the emergency air storage tanks into the eductor. The pressure at the outlet of the valve is controlled via a two-stage, self-contained pressure control operator. A failure of either stage of the pressure-regulating valve will not cause the valve to fail completely open. A failure of the second stage of the pressure-regulating valve will increase flow from the emergency air storage tanks. There is adequate margin in the emergency air storage tanks such that an operator has time to isolate the line and manually actuate the alternative delivery line.

To reduce the overall noise, the silencers, eductor, and filtration unit must all be located behind the main control area. The applicant added an ITAAC to DCD Tier 1, Section 2.2.5, to verify that the noise levels in the MCR remain below the recommended guidelines in NUREG-0700, "Human-System Interface Design Review Guidelines," issued May 2002. The ITAAC will verify that the noise level at the operator work station will remain below 65 decibels when the VES is in operation.

A filtration flow of at least 17.0 m^3/min (600 cfm) resolves the original difficulty in having a control room design in-leakage that could be demonstrated through testing. At a filtration flow rate of 17.0 m^3/min (600 cfm), 0.425 m^3/min (15 cfm) is an acceptable MCR in-leakage to maintain operator dose rates below the required levels. (The total flow at the outlet of the filters must be at least 17.0 m^3/min (600 cfm) plus the flow from the compressed air tanks.) This allows the dose analysis to assume a constant 0.142 m^3/min (5-cfm) in-leakage through the vestibule and up to 0.283 m^3/min (10 cfm) of in-leakage from sources other than through the vestibule. Using a tracer gas test, it would be possible to verify that the in-leakage from sources other than the vestibule is less than 0.283 m^3/min (10 cfm). The applicant revised the existing ITAAC related to tracer gas testing (ITTAC 7b in Section 2.2.5-5 of the DCD) to reflect the appropriate acceptance criteria of less than or equal to 0.283 m^3/min (10 scfm). A technical specification has been added to the AP1000 technical specifications to incorporate the requirements of TSTF-448 for a CRE habitability program.

The staff raised a number of technical issues in its review of the passive filter system, as described below.

6.4.2.2.1 Evaluation of Issues Associated with the Eductor in the Passive Filtration Line

Nuclear power plant applications have limited operational and maintenance experience with the eductor. The frequency of the technical specifications surveillance test of the eductor should be based on experience with eductor system degradation. However, the frequency chosen for the surveillance was not supported by a technical rationale or data on the degradation of eductors. Therefore, in RAI-SRP6.4-SPCV-09, the staff asked the applicant to justify the surveillance frequency with a technical rationale that is based on data associated with eductor degradation.

The applicant responded in a letter dated December 9, 2009, stating that the "frequency of Technical Specification Surveillance Testing for the Main Control Room Emergency Habitability System (VES) eductor was chosen to align with the Ventilation Filter Testing Program (VFTP) identified in Surveillance Requirement 3.7.6.11 of the AP1000 Technical Specifications as

revised by RAI-SRP-6.4-SPCV-06." Additionally, the applicant provided examples of operational data on eductors in industrial applications. This information supports the applicant's claims, and the staff finds the applicant's proposed TS surveillance testing frequency for the eductor reasonable.

6.4.2.2.2 Evaluation of Issues Associated with the HEPA Filter in the Passive Filtration Line

Section 6.3, regarding HEPA filter in-place leak testing, of RG 1.52 shows the acceptable combined penetration and leakage (or bypass) to be less than 0.05 percent of the challenge aerosol. The applicant's proposed TS 5.5.13 shows this value to be 0.5 percent. In its response to RAI-SRP6.4-SPCV-06 received in a letter dated May 4, 2009, the applicant stated that each HEPA filter cell is individually shop-tested to verify an efficiency of at least 99.97 percent in accordance with ASME AG-1, Section FC. The staff asked the applicant to provide a technical basis to credit 99.97 percent HEPA filter efficiency at 0.5 percent penetration and system bypass conditions.

The applicant responded in a letter date May 24, 2010 that the markup of the TS that indicates the combined penetration and leakage of less than 0.5 percent is an editorial error. The applicant corrected TS 5.5.13 in the draft DCD revision pages to indicate a leakage value of less than 0.05 percent.

In accordance with Section 6.3 of RG 1.52, to be credited with 99 percent removal efficiency for particulate matter in accident dose evaluation, a HEPA filter bank should demonstrate an aerosol leak test result of less than 0.05 percent of the challenge aerosol. It its response to RAI-SRP6.4-SPCV-06, the applicant stated that the HEPA filters will remove 99 percent of particulates consistent with the guidance in RG 1.52. The staff asked the applicant to provide a technical basis to credit 99 percent HEPA filter efficiency for particulate matter in accident dose evaluation at 0.5 percent penetration and system bypass conditions.

In a letter dated February 25, 2010, the applicant responded to RAI-SRP6.4-SPCV-10, Revision 2, by stating that it intends to comply with Section 6.3 of RG 1.52, Revision 3, as indicated in the markup of Appendix 1A in RAI-SRP6.4-SPCV-06, Revision 0. The staff found the DCD changes provided in the response to RAI-SRP6.4-SPCV-15, Revision 1 to be in alignment with the guidance and acceptable.

Section 6 of RG 1.52, Section 9.5 of ASME N510-2007, "Testing of Nuclear Air Treatment Systems," and Section 5.7 of ASME N511-2007, "In-Service Testing of Nuclear Air Treatment, Heating, Ventilating, and Air-Conditioning Systems," specify dP testing across the HEPA filter bank. However, proposed TS 5.5.13 did not specify the dP testing across the HEPA filter bank. The staff asked the applicant to explain why the technical specifications did not specify the rationale for the dP testing across the HEPA filter bank.

The applicant responded in a letter dated May 24, 2010 in RAI-SRP6.4-SPCV-15, Revision 1 that it would include dP testing across the combined HEPA filter, charcoal adsorber, and the postfilter in the ventilation filter testing program. The staff verified that the draft DCD revisions included this. As a result, the staff finds the applicant's approach acceptable.

6.4.2.2.3 Evaluation of Issues Related to the Adsorber in the Passive Filtration Line

Section 6.4, regarding adsorber in-place leak testing, of RG 1.52, Revision 3, shows the acceptable combined penetration and leakage (or bypass) to be less than 0.05 percent of the

challenge gas. The applicant's response to RAI-SRP6.4-SPCV-15, provided in a letter dated February 25, 2010, proposed TS 5.5.13, which shows this value to be 0.5 percent. In its response to RAI-SRP6.4-SPCV-06, the applicant stated that the charcoal adsorber is designed, constructed, qualified, and tested in accordance with ASME AG-1 and RG 1.140, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," Revision 2, issued June 2001. Both RG 1.52 and RG 1.140 specify a combined penetration and leakage (or bypass) adsorber in-place leak test criterion of 0.05 percent or less of the challenge gas. The staff asked the applicant to provide the technical basis for the exception taken to relax the adsorber penetration and system bypass criterion from 0.05 percent to 0.5 percent.

The applicant responded to RAI-SRP6.4-SPCV-11, Revision 1 in a letter dated December 3, 2009, by stating that it intends to comply with Section 6.3 of RG 1.52, Revision 3, as indicated in the markup of Appendix 1A in RAI-SRP6.4-SPCV-06. The applicant noted an editorial error in the markup of the technical specifications that indicates combined penetration and leakage of less than 0.5 percent. The applicant corrected TS 5.5.13 in Reference 13 to indicate a leakage value of less than 0.05 percent. As a result, the staff finds the approach acceptable.

For maximum assigned credit for active carbon decontamination efficiencies of 95 percent (elemental iodine and organic iodide), RG 1.52, Revision 3, Section 7, regarding laboratory testing of charcoal samples, shows an acceptable penetration of less than 2.5 percent for a 5.08 cm (2-in)-deep charcoal bed. For maximum assigned credit for active carbon decontamination efficiencies of 99 percent (elemental iodine and organic iodide), Section 7 shows an acceptable penetration of less than 0.5 percent for a 10.16 cm (4-in) bed. In its response to RAI-SRP-6.4-SPCV-06, the applicant stated that the charcoal filters would remove 90 percent of the elemental iodine and 30 percent of the organic iodine, claiming to be consistent with RG 1.52, Revision 2, issued March 1978. For the assigned activated carbon decontamination efficiencies of 90 percent (elemental iodine) and 30 percent (organic iodide), RG 1.52, Revision 2, Section 6, related to laboratory testing criteria for activated carbon, shows an acceptable laboratory testing criterion for a methyl iodide penetration of less than 10 percent for a 5.08 cm (2-in)-deep charcoal bed. The applicant's proposed TS 5.5.13 shows a value of 35 percent. The 35-percent allowable penetration should be calculated from a safety factor of 2 recommended by GL 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal," dated June 3, 1999, (100 percent – organic iodide efficiency)/safety factor = (100 percent – 30 percent)/2 = 35 percent). The staff asked the applicant to provide the technical basis for assigning a credit for active carbon decontamination efficiencies of 90 percent (elemental iodine) and 30 percent (organic iodide) charcoal carbon efficiency at 35 percent penetration conditions.

The applicant responded to RAI-SRP6.4-SPCV-11, Revision 1 by stating that the "technical basis for assigning activated carbon decontamination efficiencies of 90 percent (elemental iodine) and 30 percent (organic iodine) for charcoal carbon efficiency at 35 percent penetration conditions should have been identified as Reference 1." Reference 1 identifies a methodology described in GL 99-02 to calculate the allowable penetration percentage based on assumed organic iodine efficiency and a defined safety factor. Using the provided methodology, a 35 percent penetration condition is calculated assuming a 30 percent organic iodine efficiency and a safety factor of 2. The staff noted that the values in the table were generated for specific penetration, elemental and organic efficiencies, and residence times. The use of the 35 percent penetration value was not specifically approved by the staff in the RG and further justification would be needed to apply the methodology.

The applicant stated that it would revise TS 5.5.13 to show an in-place test of the charcoal adsorber with a penetration and system bypass less than 0.05 percent (RAI-SRP6.4-SPCV-10, Revision 2). Additionally, the applicant included the residence time as a design parameter in the DCD. RG 1.52, Revision 3, specifies the value of less than 0.05 percent. The applicant demonstrated that the value chosen for penetration relative to the efficiencies used in the dose analysis is conservative. The staff verified that the applicant included the change in the response to RAI-SRP6.4-SPCV-15, Revision 1 in the draft DCD revisions and finds the approach acceptable.

Section 6 of RG 1.52, Section 10.5 of ASME N510-2007, and Section 5.8 of ASME N511-2007 specify dP testing across adsorber banks. However, proposed TS 5.5.13 did not specify the dP testing across adsorber banks. The staff asked the applicant to provide the rationale for not specifying the dP testing across the charcoal filter bank in the TS.

The applicant responded to RAI-SRP6.4-SPCV-15, Revision 1 by stating that it would include dP testing across the combined HEPA filter, charcoal adsorber, and postfilter in the ventilation filter testing program. The staff verified that this was included in the draft DCD revisions. As a result, the staff finds the applicant's approach acceptable.

The staff noted in Section 6.4.2.3 of NUREG-1793 that the applicant referenced RG 1.140 for the design, construction, and qualification of the charcoal adsorber. The staff notes that it cited RG 1.52, Revision 3, for all other aspects of the design. The staff is unclear why the applicant would use RG 1.140 for this specific aspect of the application rather than RG 1.52, Revision 3. In a letter dated May 24, 2010, the applicant revised its response to RAI-SRP6.4-SPCV-15 to state that RG 1.52, Revision 3, is the correct reference and provided an associated DCD change. The staff found the response acceptable.

6.4.2.2.4 Evaluation of the Test Frequency of the Combined Filters Pressure Drop

TS 5.5.13, which the applicant submitted in response to RAI-SRP6.4-SPCV-15, Revision 0, states that a "program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in accordance with regulatory guide 1.52, Revision 3, ASME N510-1989, and AG-1." Item d of TS 5.5.13 requires a test of "the pressure drop across the combined HEPA filter, the charcoal adsorber, and the post filter." However, RG 1.52, Revision 3, ASME N510-1989, and AG-1 do not have a clearly defined combined pressure drop test frequency. This dP measure is a surveillance test made at regular intervals to detect deterioration that may develop under service conditions. Regular in-place testing is necessary because deterioration may take place even when the system is not being operated.

For the combined HEPA filter, charcoal adsorber, and postfilter pressure drop test in TS 5.5.13.d, the staff asked the applicant to provide a specific citation and reference for performing this test and to provide the required test frequency. The applicant responded in a letter dated June 2, 2010, to RAI-SRP6.4-SPCV-16 by stating that it would revise the proposed TS 5.5.13 to list the frequencies for each test listed in TS 5.5.13.a, b, c, and d. It removed the reference to RG 1.52, Revision 3, and ASME N510-1989 from TS 5.5.13.d that were proposed in the response to RAI-SRP6.4-SPCV-15. The frequencies provided for Section 5.5.13.a, b, and c are the same as the frequencies listed in RG 1.52, Revision 2. As a result, the staff finds these frequencies acceptable. The test in TS 5.5.13.d will be conducted every 24 months, which aligns with the frequencies for the HEPA filter and charcoal adsorber inplace tests and

the charcoal adsorber sampling and analysis (TS 5.5.13.a, b, and c). A specific RG or standard citation is not needed with the revised TS 5.5.13.d, since the test is described by the TS itself and the frequency is specified.

Because the frequencies given in TS 5.5.13.a, b, and c are those specified in RG 1.52 and the frequency given in TS 5.5.13 d is aligned with the frequencies in the RG, the staff finds the DCD changes identified in the response to RAI-SRP6.4-SPCV-16 acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

6.4.2.2.5 Evaluation of the Safety Class of Passive Filtration Flow Instrumentation

Redundant flow instruments are located downstream of the filtration units to ensure that adequate flow is passing through the filtration units during testing activities. The instrumentation is not safety-related. The applicant's rationale is that the instrument does not perform a safety function. An existing VES safety flow instrument indicates whether there is sufficient flow coming from the compressed air tanks to induce the passive filtration flow. The new flow instrumentation has high and low alarms to alert the operator of possible filtration issues during testing.

Section 3.3.1.3 of ANSI/ANS-51.1-1983 (Reference 2) states the following:

> safety class 3 (SC-3) shall apply to equipment, not included in SC-1 or -2, that is designed and relied upon to accomplish the following nuclear safety functions:
>
> > k. Ensure nuclear safety functions provided by SC-1,-2, or -3 equipment, m. Provide information or controls to ensure capability for manual or automatic actuation of nuclear safety functions required of SC-1, -2, or -3 equipment"

At a filtration flow rate of 17.0 m^3/min (600 cfm), 0.425 m^3/min (15 cfm) is an acceptable MCR in-leakage level to maintain operator dose rates below the required levels. (The total flow at the outlet of the filters must be at least 17.0 m^3/min (600 cfm) plus the flow from the compressed air tanks.) This allows the dose analysis to assume a constant 0.142 m^3/min (5-cfm) in-leakage through the vestibule and up to 0.283 m^3/min (10 cfm) of in-leakage from sources other than through the vestibule. The 17.0 m^3/min (600 cfm) flow induced by the eductor is the design basis used by the dose calculation to make sure filtration units in the passive air filtration line will work to remove particulate and iodine from the air to reduce the potential MCR dose. This dose calculation is required to satisfy GDC 19 in Appendix A to 10 CFR Part 50.

The two flow instruments in the filtration line provide information to ensure the capability of the eductor to draw at least 17.0 m^3/min (600 cfm) so the VES system safety function (MCR habitability during radiological accidents) can be achieved. The existing VES safety flow instrumentation to indicate whether there is sufficient flow (1.84 m^3/min (65 cfm)) coming from the compressed air tanks to induce the passive filtration is not a direct indication of the performance of the eductor.

The operators would rely on this instrumentation during an accident to ensure that the safety-related filtration train was functioning. Based on ANSI/ANS-51.1-1983, the staff asked that at least one flow instrument in the passive air filtration line be safety related. The staff also

asked the applicant to provide additional justification that the operators would not rely on this instrumentation during an accident or else make one of the instruments safety related.

The applicant responded in a letter dated February 25, 2010, to RAI-SRP6.4-SPCV-15, Revision 0, by stating that the operator would not rely on this instrumentation during an accident. Rather, the operator would rely on the flow instrumentation directly from the compressed air tanks. The staff noted that this instrumentation would not notify the operator of a clogged or partially blocked component in the filter train and that flow blockage was a credible failure after 24 hours. The applicant responded by making design changes demonstrating compliance with the single failure criteria. Further discussion of the single failure criteria appears below. Given that the applicant eliminated all credible single failures, the staff agrees that reliance on the flow instrumentation from the compressed air tanks during an accident is acceptable. As a result, the two redundant flow instruments that are located downstream of the filtration units to ensure that adequate flow is passing through the filtration units during testing activities do not need to be safety-related.

6.4.2.2.6 Evaluation of the Single Failure of the Passive Filtration Line

The applicant stated that redundant passive filtration lines are not required because the passive filtration line has no electrical power requirements, contains no moving parts, and requires no maintenance such as adjusting setpoints or lubricating bearings. The applicant stated that adequate design margin is provided to prevent the likelihood of a passive failure.

Section 3.2.1.c of ANSI/ANS-51.1-1983 states that fluid systems required to support, directly or indirectly, the three nuclear safety functions stated above shall be capable of performing these functions as provided in ANSI/ANS-58.9-1981, "American National Standard Single Failure Criteria for Light Water Reactor Safety-Related Fluid Systems."

ANSI/ANS-58.9-1981 defines "passive failure" as "a failure of a component to maintain its structural integrity or the blockage of a process flow path." In this standard, the term refers to a random failure, its consequential effects assumed in addition to an initiating event, and its consequential effects for the purpose of safety-related fluid system design and analysis. This standard defines rules for application of the single failure criteria. During the short term, the single failure considered may be limited to an active failure. During the long term, assuming no prior failure during the short term, the limiting single failure considered can be either active or passive. "Long term" is defined as that period of safety-related fluid system operation following the short term, during which the safety function of the system is required. "Short term" is defined as that period of operation up to 24 hours following an initiating event.

Additionally, TSTF-448 and the associated TS Bases describe the expectation to consider passive failures. Specifically the TSTF states, "No single active or passive failure will cause the loss of outside or recirculated air from the CRE [control room envelope]."

The VES passive filtration line is required to meet single failure criteria. The single active or passive failure of a component in the VES passive filtration line, assuming a loss of outside power, shall not impair the ability of the system to perform its design function. The staff noted that if the present passive filtration design proposed by the applicant has a passive failure (e.g., the nozzle section of the eductor fails to induce the minimum 17.0 m^3/min (600 cfm) flow), the safety function of the filtration may not be achievable. The proposed VES passive filtration line does not have independent, redundant trains to recirculate and filter the CRE. The staff asked the applicant to provide a justification that the described system meets the single failure

criteria or to provide a redundant filter train. The applicant responded to RAI-SRP6.4-SPCV-13, Revision 0 in a letter dated December 11, 2009 by stating the following:

> The primary components that comprise the main control room habitability system (VES) passive filtration line are duct work, two silencers, an eductor, and a filtration unit. The passive filtration line has no active components. The only active components in the VES are in the air delivery portion of the system that provides the motive flow to induce to the filtration flow. The air delivery portion of the system also provides breathable air for main control room occupants during abnormal scenarios. In the air delivery portion of the system, there are redundant flow paths. The redundant flow paths prevent a single active or passive failure from impairing the ability of the system to perform its design function. Based on the guidance in SECY-77-439, the passive filtration portion of the system must be evaluated for a credible passive failure 24 hours after the start of an event. SECY-77-439 defines a passive failure as events such as a line blockage or structural failure of a static component that limits the effectiveness of the component. Though a passive failure in the passive filtration portion of the VES is highly unlikely, it would not impair main control room habitability. Dose analysis for the AP1000 main control room was performed to verify that in the event of a passive failure in the passive filtration portion of the VES 24 hours after the initiation of the event operator doses would remain below 5 rem TEDE. The limiting AP1000 main control room dose scenarios were evaluated for a loss of filtration flow 24 hours into an accident. These scenarios are limiting since they involve a release 24 hours after the initiation of the event. The analysis showed the following acceptable increases in dose rates compared to the scenarios when filtration is available for 72 hours. Therefore, the passive filtration portion of the VES can sustain a single passive failure without impairing main control room habitability for the first 72 hours following a design basis accident.

The applicant demonstrated that filtration was not needed in the long term or after 24 hours. The staff noted that this response adequately addressed the blockage of the filter or ductwork. However, it did not address the potential blockage of the eductor, which would prevent the required breathable air from entering the control room. These concerns were brought to the attention of the applicant. In a letter dated February 25, 2010, the applicant submitted a response to RAI-SRP6.4-SPCV-13, Revision 2, which incorporate a bypass line for the eductor and thereby provides operators with the ability to deliver the air flow from the emergency air storage tanks following the highly unlikely passive single failure of the eductor. The design addition added a flow control orifice and normally closed manual valve to allow the 65 scfm of breathable air to flow into the MCR without passing through the eductor. Also, the applicant added a normally open manual valve upstream from the eductor to provide full isolation of the flowpath to the educator. The revised dose analysis demonstrates that air filtration is not necessary in the long term. The manual bypass valves with the safety-related flow meter from the compressed air tanks allow the air delivery function to be accomplished if there is a blockage. As a result, the staff finds the applicant's response acceptable because the design is capable of withstanding the unlikely credible single failures in the passive filter train.

The applicant has designed a filter system that meets, to the extent practicable, the guidance of RG 1.52. The applicant proposed surveillance and testing consistent with the intent of RG 1.52 and the STS. The design was demonstrated to function in scale-model testing. The applicant

has addressed all credible single failures. As a result, the staff finds the single filter train acceptable.

6.4.2.3 Evaluation of Design Changes To Reduce Unfiltered In-Leakage

The applicant changed the MCR envelope purge design. The VES discharge air flow is now directed into the entry vestibule to provide a continuous vestibule purge. This helps to reduce the radioactivity introduced into the MCR each time there is access to or from the MCR during a radiological accident. The VES discharge dampers originally discharged through the MCR wall directly to the atmosphere outside the MCR. This design change redirects the damper discharge flow into the MCR vestibule and adds openings to allow free passage from the vestibule to the hallway.

The applicant made a number of other changes to the design of the CRE. For example, the applicant eliminated ductwork penetrating the CRE. The applicant also installed isolation capability for various control room penetrations, like the sanitary system and normal vents. Additionally, the applicant proposed different materials for the control room. Each of these changes improves the overall design, and the staff finds them acceptable. The staff notes that the overall effectiveness of the CRE is demonstrated through the testing associated with the control room integrity program.

During the course of the review, the applicant proposed a number of other changes. For example, the applicant proposed an additional actuation to isolate the normal control room ventilation system. The applicant also proposed to remove the dose calculations associated with the normal ventilation system. According to TR-122, "AP1000 COL Standard Technical Report Submittal of APP-GW-GLN-122 (TR-122)," Revision 0, dated July 27, 2007; the DCD did not report the dose analysis for the nuclear island non-radioactive ventilation system (VBS) operating cases because the VBS does not continue to operate if a high-2 radioactivity level is detected. The staff was concerned that the normally operating active ventilation system (VBS) in the supplemental air filtration mode should not be isolated early and should be demonstrated to meet the regulatory limits. With active systems, the VBS is the preferable system to function because it will provide greater comfort to the operators and will provide active cooling for the control room equipment. In response to RAI-TR122-SPCV-01, the applicant stated that it will update the DCD to present the results for the case with the VBS operating for the duration of the accident and remove the earlier isolation of the VBS.

The applicant also proposed to procedurally delay access into the control room. According to TR-122, Revision 0, the AP1000 requires procedurally delaying access to the control room in order to meet the operator dose limits. The staff was concerned that GDC 19 requires that adequate radiation protection shall "be provided to permit access and occupancy of the control room under accident conditions." The AP1000 control room, as proposed by TR-122, did not seem to provide adequate radiation protection to permit access under accident conditions because of the access control process. In its response to RAI-TR122-SPCV-01, submitted in a letter dated December 17, 2008, the applicant stated that it would update the DCD to present the results for the case without consideration for an MCR entry time delay. The results will show that the AP1000 design with the vestibule purge configuration fully satisfies GDC 19, with or without credit for the use of the entry time delay. The staff finds this response acceptable.

6.4.2.4 Redesignation of Technical Support Center

The applicant proposed to redesignate what was referred to as the TSC in the certified design to the control support area. The change removed the TSC from the Section 6.4 description in Tier 2. If a reactor is being built on a site that already has an operating reactor and a functioning TSC, it is reasonable for the COL applicant to want a single TSC. The applicant added a statement to Tier 2* that requires the TSC to be located in the control support area. As a result, if a COL applicant would like to locate the TSC somewhere other than in the control support area, a Tier 2* departure requiring NRC approval would be necessary. In a subsequent letter dated January 27, 2010, the applicant removed the Tier 2* information and included a statement in DCD Chapter 18 indicating that the location of the TSC would be in the control support area. Section 13.3 of this report approves this change. Because the information is Tier 2, rather than Tier 2*, a departure may not require prior NRC approval. However, the location of the TSC is part of the Emergency Plan, which a COL application is required to include under 10 CFR 52.79(a)(21), "Contents of applications; technical information in final safety analysis report." As a result, regardless of whether the Tier 2 departure would require NRC approval, the Emergency Plan and the location of the TSC therein would be subject to prior NRC review and approval. The analysis showing that the control support area remains habitable following an accident, if the nonsafety-related VBS system is available, remains in the DCD. The analyses were redone in the amendment with the higher dispersion factors, and the results remain within regulatory limits. Although these values were removed from the DCD in one of the intervening revisions, the DCD now refers to these analyses as the "VBS Operating" dose results. Because there are no impacts from a safety perspective associated with this change, the staff finds the change acceptable.

6.4.2.5 Changes to Improve Operational Flexibility

TS 3.9.7 specifies that the minimum radioactive decay time ensures that the radiological consequences of a postulated fuel handling accident inside containment or in the fuel handling area inside the auxiliary building are consistent with the assumptions in AP1000 DCD Tier 2, Chapter 15.

In TR-122, the applicant changed the minimum decay time mandated in the TS to 48 hours. This TS change affects the convenience of plant operations; there are no system design concerns. The staff finds this change acceptable based on an acceptable finding associated with the related dose calculations that are being reviewed by the Office of New Reactors, Division of Site and Environmental Reviews, Technical Specification Branch.

Supplementary changes have also been made to VES to increase the reliability of the system and provide a safe environment for workers performing maintenance on the system. Because there was concern about isolating this system for repair or maintenance with a single valve, double isolation valves have been added to the banks of air tanks and the supply lines. Individual fill and vent lines have also been added to each bank of tanks to allow one bank to be taken out of service and be recharged while keeping the other three banks of tanks online and ready for service at all times. Revised Figure 6.4-2 in the DCD depicts these changes.

The applicant modified the condition in TS 3.7.6 for "VES inoperable" to specify a condition for "One bank of VES air tanks (8 tanks) inoperable." The completion time to increase pressure in the operable tanks to the upper portion of the system operating band is 12 hours. The completion time is 7 days to restore the VES to operable status.

In the revised TS bases (B 3.7.6), the applicant stated the following:

> If one bank of VES air tanks (8 tanks out of 32 total) is inoperable, then VES is able to supply air to the MCR for 54 hours (75 percent of the required 72 hours). If VES is actuated, operator must take actions to maintain habitability of the MCR once the air in the tanks has been exhausted. The VBS supplemental filtration mode or MCR ancillary fans are both capable of maintaining the habitability of the MCR after 54 hours.
>
> Increasing the pressure in the OPERABLE tanks from the minimum pressure of 3400 psig to the upper portion of the system operating band maximizes the time the VES will be able to supply air to the MCR. The 12-hour Completion Time provides sufficient time to achieve the increased pressure.
>
> With one bank of VES air tanks inoperable, action must be taken to restore OPERABLE status within 7 days. In this Condition, the remaining OPERABLE VES air tanks, along with compensatory operator actions, are adequate to protect the main control room envelope habitability. The 7 day Completion Time is based on engineering judgment, considering the low probability of an accident that would result in a significant radiation release from the reactor core, the low probability radioactivity release, and that the remaining components and compensatory systems can provide the required capability.

The staff reviewed the rationale. At 75 percent VES air capacity (54 hours); the system no longer accomplishes the safety function for 72 hours. Additionally, the VBS supplemental filtration is not safety-related. The applicant's regulatory treatment of non-safety systems (RTNSS) (ancillary fans) may not be available in 54 hours. Additionally, increasing the pressure in the operable tanks is not a reviewed operating condition and may not be advisable with a degraded system. Further, restoring or replacing an inoperable tank is a relatively simple evolution, and it is not clear why 7 days is needed to complete this action.

The 72-hour design basis for passive safety system capability and operator actions 72 hours after accident initiation has been evaluated as part of the RTNSS process. The safety-related design basis for the VES is to operate for 72 hours. The 54-hour VES air capacity is in a condition outside the accident analysis. If one bank of VES air tanks (8 tanks out of 32 total) is inoperable, the loss of safety function would merit more immediate action.

The staff asked the applicant to explain why such a long completion time is appropriate for the loss of a safety function to restore operability.

The applicant responded in a letter dated December 3, 2009, to RAI-SRP6.4-SPCV-14, Revision 0 by stating the following:

> Any leakage that would cause the system to enter the Technical Specification of "One bank of VES air tanks (eight tanks) inoperable" would most likely be at the safety-relief valves (V040A/B/C/D). The air bank with the excessively leaking relief valve will be individually isolated and depressurized to allow the safety-relief valve to be replaced. Because only the air bank with the leaking safety-relief valve is isolated, the remaining three air banks will be at or above the minimum operating pressure of the system. The maintenance on an individual air bank can be done without affecting the other 3 banks in any way.

The pressure in the operable tanks will not be increased. The operable tanks are credited as maintaining a normal minimum pressure of 3400 psig.

The time limit of 7 days for this Technical Specification is acceptable based on engineering considerations with regard to the low probability of an accident that would result in a significant radiation release from the fuel, the low probability of not containing the radiation, and that the remaining components and compensatory systems can provide the required capability to maintain the MCRE [ ] habitable. If one bank of tanks is taken out of service, 75 percent of the system will still be available to supply air to the control room in the event of an accident. Dose calculations have been performed to verify that the MCR dose limits will remain within the requirements of GDC 19 if 75 percent (54 hour supply of breathable compressed air) of VES is available and compensatory measures, through the use of the ancillary fans, are taken at 54 hours for the remainder of the event. The MCR ancillary fans are located in the auxiliary building as indicated in Tier 1 Table 2.7.1-5 of the DCD and will be available if required 54 hours after the initiation of VES.

The Tech Spec Bases 3.7.6 as submitted in RAI-SRP6.4-SPCV-06, Revision 0 are marked up to include that GDC 19 requirements are met with 54 hours of VES with compensatory measures taken at 54 hours.

Based on discussions with the staff at a public meeting on December 15, 2009, the applicant revised this response in a letter dated March 25, 2010. RAI-SRP6.4-SPCV-14, Revision 2, included additional required actions in Limiting Condition for Operation (LCO) 3.7.6, Condition D, to provide greater confidence that the MCR envelope can be maintained habitable for 72 hours following a DBA during a condition where one bank of VES emergency air storage tanks is not available. The additional action requires confirmation that the VBS MCR ancillary fans and supporting equipment are available within 24 hours of a bank of VES emergency air tanks going out of service.

Action D.1 has also been clarified. The applicant revised action D.1 to require verification of the pressure in the unaffected banks of air tanks. The pressure in the affected tanks should be verified to be above 23.4 MPa (3,400 psig) within 2 hours and once every 12 hours thereafter. The applicant removed the reference to increasing the pressure in the operable tanks above 23.4 MPa (3,400 psig) to the upper portion of the system operating band from action D.1 and the associated bases. Based on the discussion above and an acceptable finding from the Office of New Reactors Technical Specification Branch, the staff finds the applicant's TS adequate.

6.4.3 Conclusion

The staff has reviewed the applicant's changes to the DCD on the VES system. On the basis of the evaluation described in NUREG-1793 and this report, the staff concludes that the portion of the DCD on the VES system is acceptable and that the application to amend the DC meets the requirements of Subpart B, "Standard Design Certifications," of 10 CFR Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," that are applicable and technically relevant to the AP1000 standard plant design. This conclusion is based on acceptable findings by the other responsible NRC branches. Notably, there are related changes in Chapters 3, 7, 9, 15, and 16, which are the responsibility of other branches. Additionally, the DCD revisions reviewed by the staff were submitted as draft.

In a letter dated July 29, 2010, the applicant submitted Change Notice 73. In the letter the applicant identified that the design change noted in this section required a higher pressure at the pressure regulator. As a result, the minimum pressure in the canistered air tanks also needs to be higher. The technical evaluation of this aspect of the design change is included in Chapter 23 of this report.

6.5.2 Containment Spray System

6.5.2.1 Summary of Technical Information

In Revision 17 of the DCD, the applicant changed the description of the nonsafety-related containment spray system in Section 6.5.2.1 and Figure 9.5.1, "Fire Protection System P&ID," in order to correct errors.

6.5.2.2 Evaluation

Revision 15 of DCD Tier 2, Section 6.5.2.1.1, states that the remotely operated valve (FPS-V701) downstream of the manual isolation valve in the spray riser is normally closed, but Figure 9.5.1 shows it as open. In Revision 17 of the DCD, the applicant changed the text to describe this valve as open to match Figure 9.5.1. However, as described in the applicant's January 8, 2009, response to RAI-SRP6.0-SPCV-02, Revision 1, this valve is actually normally closed; therefore, the applicant proposed changes to Section 6.5.2.1.1 and Figure 9.5.1 of the DCD to describe it as such. The staff finds the changes acceptable because the text will revert to the Revision 15 description of the valve and the original error in Figure 9.5.1, which showed the valve as open, will be corrected.

Revision 15 of DCD Tier 2, Section 6.5.2.1.1, incorrectly states that closing the passive containment cooling water system fire header isolation valve (PCS-V005) isolates the primary fire water tank, when it actually isolates the PCCWST. Revision 17 corrects this error, and the text is now consistent with Tier 1, Figure 2.3.4-1, "Fire Protection System," and Figure 6.2.2.1, "PCS P&ID."

6.5.2.3 Conclusion

These changes are acceptable and have no impact on the evaluation of the nonsafety-related containment spray system reported in NUREG-1793, Section 6.5.2.

6.6 Inservice Inspection of Class 2, 3, and MC Components

6.6.1 Summary of Technical Information

In Revision 16 to the AP1000 DCD, the applicant added metallic containment (Class MC) components to the scope of the preservice inspection (PSI) and inservice inspection (ISI) programs. Furthermore, the changes defined the responsibilities for preparation of the PSI and ISI programs and removed the ISI program discussion from Section 3.8.2, "Steel Containment," which is dedicated to the design, rather than the inspection of the containment.

In Revision 17, the applicant revised Section 6.6.2 to delete the phrases that there are no Quality Group B and C components that require ISI during operation and that relief from Section XI requirements for the baseline DC code will not be required.

6.6.2 Evaluation

In Revision 16 of the AP1000 DCD, the applicant proposed to add Class MC components to the scope of the PSI and ISI programs. 10 CFR 50.55a(b)(2)(vi), "Codes and standards," provides requirements for licensees to select their effective Edition and Addenda of Subsections IWE and IWL for ASME Code Section XI, as modified and supplemented by the requirements in paragraphs (b)(2)(viii) and (b)(2)(ix). ASME Code Section XI, Subsection IWE is dedicated to the PSI and ISI of Class MC components. The AP1000 DCD, Revision 16 changes ensure that the metallic containment integrity is maintained through periodic inspection and testing as defined under the regulations and ASME Code Section XI. In addition, the inclusion of Class MC components as part of the PSI/ISI program provides heightened visibility of the operational program by including Class MC components along with ASME Code Class 2 and 3 components, which receive PSI/ISI. The staff concludes that the change to include Class MC components within the PSI and ISI program is in compliance with the requirements of 10 CFR 50.55a and ASME Code Section XI, and is, therefore, acceptable.

In Revision 17 of the AP1000 DCD, Section 6.6.2, the applicant proposed to remove the phrase that there are no Quality Group B and C components that require ISI during operation.

In the AP1000 DCD, Section 6.6.2, the applicant states that ASME Code Class 2, 3, and MC components are designed so that access is provided in the installed condition for visual, surface and volumetric examinations specified by the ASME Code. Furthermore, it states that considerable experience has been used in designing, locating, and supporting Quality Group B and C (ASME Code Class 2 and 3) pressure-retaining components to permit PSI and ISI. The applicant's removal of the statement that there are no Quality Group B and C components that require ISI during reactor operation does not eliminate the performance of ISI. Furthermore, neither 10 CFR 50.55a nor ASME Code Section XI prohibits the performance of ISI during plant operation. Since the proposed changes would allow the possibility to expand the performance of ISI during operational conditions rather than eliminate the ISI examinations, the staff concludes that this portion of the proposed change is acceptable.

In Revision 17 of the AP1000 DCD, Section 6.6.2, the applicant proposed to remove the sentence: "Relief from Section XI requirements will not be required for ASME Section III, Class 2, 3, and MC pressure retaining components in the AP1000 plant for the baseline design certification code."

In the AP1000 DCD, Section 6.6.2, the applicant states that ASME Code Class 2, 3, and MC components are designed so that access is provided in the installed condition for visual, surface and volumetric examinations specified by the ASME Code. Furthermore, it states that considerable experience has been used in designing, locating, and supporting Quality Group B and C (ASME Code Class 2 and 3) pressure-retaining components to permit PSI and ISI. The AP1000 DCD also states that the goal of designing for inspectability is to provide for the inspectability, access and conformance of component design with available inspection equipment and techniques. Factors such as examination requirements, examination techniques, accessibility, component geometry and material selection are used in evaluating component designs, as stated by the applicant.

In addition, the regulations under 10 CFR 50.55a(g)(3)(ii) require that Class 1, 2, and 3 components and supports be designed and provided with access to enable the performance of both PSI and ISI requirements set forth in the Editions and Addenda of Section XI of the ASME

Code incorporated by reference in paragraph (b) of this section applied to the construction of the particular component. However, 10 CFR 50.55a recognizes that inaccessible areas for the containment cannot be eliminated completely. 10 CFR 50.55a(b)(2)(ix)(A) states that for Class MC applications, the licensee shall evaluate the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of or result in degradation to such inaccessible areas. The staff concludes that proposed changes recognize that inaccessible areas may be present in the design of Class MC components and that sufficient effort will be incorporated into the design of Class 1, 2, and 3 components to meet the requirements under 10 CFR 50.55a. The staff concludes that the change is in compliance with the regulations and is, therefore, acceptable.

6.6.3 Conclusion

Based on the above evaluation, the staff concludes that the AP1000 DCD changes as proposed in Revisions 16 and 17 meet the requirements of 10 CFR 50.55a and ASME Code Section XI for Class 2, 3, and MC components and are, therefore, acceptable. Therefore, the proposed DCD changes are acceptable pursuant to 10 CFR 52.63(a)(1)(vii) on the basis that they contribute to the increased standardization of the certification information.

7. INSTRUMENTATION AND CONTROL

Westinghouse (the applicant) has submitted information in support of its design certification (DC) amendment application that the applicant considers "proprietary" within the meaning of the definition provided in Title 10 of the *Code of Federal Regulations* (10 CFR) 2.390(b)(5), "Public inspections, exemptions, requests for withholding." The applicant has requested that this information be withheld from public disclosure and the Nuclear Regulatory Commission (NRC) staff agrees that the submitted information sought to be withheld includes proprietary commercial information and should be withheld from public disclosure. This chapter of the NRC staff's evaluation includes proprietary information that has been redacted in order to make the evaluation available to the public. The redacted information appears in "square brackets" as follows:

[ ]

The complete text of this chapter, including proprietary information, can be found at Agencywide Documents Access and Management System (ADAMS) Accession Number ML112091879 and can be accessed by those who have specific authorization to access Westinghouse proprietary information.

7.1 Introduction

Chapter 7 of the AP1000 design control document (DCD) includes changes to the descriptions of commitments pertaining to the primary instrumentation and control (I&C) systems of the AP1000 design, as evaluated in NUREG-1793, "Final Safety Evaluation Report Related to the Certification of the AP1000 Standard Design," issued by the NRC in September 2004. Additionally, AP1000 DCD Tier 1, Section 2.5, includes changes to the proposed design description and inspections, tests, analyses, and acceptance criteria (ITAAC) for I&C systems. This report must be used in concert with the original version of NUREG-1793 and Supplement 1 to completely understand the full evaluation of the AP1000 I&C systems standard design. The sections identified and addressed below have had additions, alterations, or deletions incorporated into the technical information presented previously in the certified design of Revision 15 of the AP1000 DCD. The sections not listed below had no appreciable technical changes and, thus, are not included in this report.

Although all open items have been resolved prior to the final issuance of this supplement of the safety evaluation report (SER), this report discusses what open items were generated when it considered an applicant's original submittal and response to be inadequate.

In the proposed changes to the AP1000 DCD, the applicant provided additional information related to the architecture of its safety-related I&C protection system, referred to as the protection and safety monitoring system (PMS); the diverse back-up system to the PMS, the diverse actuation system (DAS); and data communications protocols and methods utilized to ensure a secure development and operational environment (SDOE). The applicant also proposed minor modifications to some of the interlock and control systems.

7.1.3.1 Compliance with Standard Review Plan (SRP) Criteria

The staff reviewed the additional and amended information provided by the applicant, using the guidance in Chapter 7, "Instrumentation and Control Systems," of NUREG-0800, "Standard

Review Plan [SRP] for the Review of Safety Analysis Reports for Nuclear Power Plants: Light Water Reactor Edition," Revision 5. The NRC developed NUREG-1793 using the guidance of Revision 4 of NUREG-0800, which did not include Section 7.8, "Diverse Instrumentation and Control Systems," or Section 7.9, "Data Communication Systems." Therefore, although this supplement follows the format of NUREG-1793, the NRC added Section 7.8, "Diverse Instrumentation and Control Systems," and Section 7.9, "Data Communication Systems," to discuss the staff's review of these issues. Where necessary, the staff correlated the information in Sections 7.8 and 7.9 with other pre-existing Chapter 7 sections.

7.1.3.2 Compliance with Industry Standards

The applicant submitted a number of technical reports (TR) associated with I&C systems, which it incorporated by reference into the AP1000 DCD. Based upon the letter, "Secondary References in a Design Certification Rule," dated May 3, 1994, the staff determined with the concurrence of utility representatives and other members of the commercial nuclear industry that documents referenced in Tier 1 and Tier 2 information in the AP1000 DCD should be considered part of the licensing basis. Therefore, the addition of these documents is deemed acceptable to the staff.

In several cases, the applicant referenced revisions of the same industry standards and other regulations in its TRs different from those referenced in DCD Revision 15. The applicant stated that all newly created or revised technical documents that do not refer to the guidance or standards certified in Revision 15 of the AP1000 DCD will reference currently issued guidance or standards provided the newly referenced guidance or standards include all acceptance criteria to those references certified in the Revision 15. The staff found the response acceptable.

7.1.3.3 Compliance with 10 CFR Part 52

Based upon the discussion in Section 7.1.3.3 of NUREG-1793, the standardized power plant designer or combined license (COL) applicant will satisfactorily demonstrate that "the digital I&C system design development process, as documented in the DCD, will ensure that the digital I&C system, as designed, will satisfactorily accomplish its safety functions."

All newly submitted documentation is to support and agree with all previously submitted design documents to enable the NRC to reach the same conclusion as in NUREG-1793.

The regulations in 10 CFR 52.63, "Finality of standard design certifications," are applicable to modifications addressed in this chapter, which supplements NUREG-1793 to address the applicant's amendment request.

7.1.4 Tier 1 Material

Revision 19 of AP1000 DCD Tier 1, Section 2.5, includes information associated with the DAS and the PMS within the first two phases of the software, logic-based, or programmable technology design processes. The proposed alteration to the DAS design allows the use of technology other than microprocessor-based systems (i.e., firmware or analog technology) and the selection of the specific PMS design platform, the Common Q platform. Revision 19 of the DCD also proposes to remove part of the design acceptance criteria (DAC), which is a subset of ITAAC, in Tier 1, Section 2.5. Sections 7.2 and 7.8 of this report discuss these changes.

The applicant classifies the first two phases of the PMS and DAS lifecycles as the design requirements (or conceptual) phase and system definition phase, for both the PMS and the DAS in the certified design. Branch Technical Position (BTP) 7-14, Revision 5 as well as Branch Technical Position HICB-14, to which the AP1000 DCD was certified, in Chapter 7 of NUREG-0800 refers to these two phases of development as the "Planning Activities" phase and the "Requirements Activities" phase of the PMS software lifecycle (SLC) and the DAS programmable technology lifecycle, respectively.

AP1000 DCD Tier 1, Section 2.5.1, identifies the ITAAC for the DAS, and Section 2.5.2 identifies the ITAAC for the PMS. Section 2.5.1, Design Description 4, and Table 2.5.1-4, "Inspections, Tests, Analyses and Acceptance Criteria," discuss Design Commitment 4, which identifies the phases of the programmable technology development life cycle for the DAS. Likewise, Section 2.5.2, Design Description 11, and Table 2.5.2-8, "Inspections, Tests, Analyses and Acceptance Criteria," discuss Design Commitment 11, which identifies all phases of the SLC for the PMS. Based on the cover letter received with Revision 17 of the AP1000 DCD, the applicant considers both of these items DAC.

To address the DAC, the applicant provided design information related to the design requirements and system definition phases of the PMS SLC and the DAS programmable technology lifecycle. For both the DAS and PMS sections (Sections 2.5.1 and 2.5.2, respectively) of the Tier 1 document, the applicant proposed the removal of the first two phases of the SLC processes in Revision 17 of the AP1000 DCD. Sections 7.2.5 and 7.2.8 of this report provide further discussion of this topic as it relates to the PMS, and Sections 7.8.2 and 7.8.3 discuss the lifecycle development process for the DAS.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

7.1.5 Instrumentation and Control System Architecture

In support of the proposed changes to the AP1000 DCD, the staff reviewed the following TRs:

- APP-GW-GLR-071/ Westinghouse Commercial Atomic Power (WCAP)-16675-P, "AP1000 Protection and Safety Monitoring System Architecture Technical Report," Revision 5 (TR-89). This report describes how the PMS will function. Section 7.2.2, "Protection and Safety Monitoring System Description," and Section 7.9, "Data Communications Systems," discuss this report.

- APP-GW-GLR-065/WCAP-16674-P, "AP1000 I&C Data Communication and Manual Control of Safety Systems and Components," Revision 4 (TR-88). This report provides critical design aspects of the communications methodology and various protocols when dealing with inter- and intra-division communications and safety-related to nonsafety-related communications methods. The report also includes key information related to the manual operation of the AP1000 safety systems. Sections 7.2.2, 7.5.3, 7.9.3, and 7.9.4 of this report discuss WCAP-16674-P in greater detail.

- APP-GW-GLN-022, "AP1000 Standard Combined License Technical Report DAS Platform Technology and Remote Indication Change," Revision 1 (TR-97), dated May 2007. This report provides information associated with the relocation of DAS equipment, and it also incorporates changes to allow a microprocessor-based or

alternative technology to serve as the principal design of the DAS platform. Section 7.1.6, "Diversity and Defense in Depth Assessment of the AP1000 Protection System," and Section 7.8, "Diverse Instrumentation and Control Systems," of this report provide additional discussion on APP-GW-GLN-022.

- APP-GW-GLR-017, "AP1000 Standard Combined License Technical Report," Revision 0 (TR-42). This report summarizes the applicant's proposed resolution of the 10 generic open items (GOIs) and 14 plant-specific action items (PSAIs) associated with the NRC review of the Westinghouse Common Q platform. Section 7.2.3 includes more information on this document.

- APP-GW-GLR-024/WCAP-16361-P, "AP1000 Setpoint Calculations for Protective Functions," Revision 1 (TR-28). This report discusses the calculation of setpoints and setpoint methodology for the PMS. Section 7.2.7 addresses additional information on setpoint methodology in AP1000 I&C systems.

- APP-GW-GLR-018, "Failure Modes and Effects Analysis and Software Hazards Analysis for AP1000 Protection System," Revision 0 (TR-43). This report summarizes the steps taken to perform the failure modes and effects analysis (FMEA) and software hazards analysis (SHA) and serves primarily as a pointer to the AP1000 FMEA and SHA reports.

- APP-GW-JJ-002/WCAP-16438-P, "FMEA of AP1000 Protection and Safety Monitoring System," Revision 3. This report provides the postulated failure modes and effects the PMS will undergo as a result of the given failures.

- APP-PMS-GER-001/WCAP-16592-P, "Software Hazard Analysis of AP1000 Protection and Safety Monitoring System," Revision 2. This report discusses the risks associated with the use of software or programmable technology in protection and control systems. The report will be discussed further in Section 7.2.

- APP-GW-GLN-004, "Instrument and Control Design Change," Revision 0 (TR-39), incorporates signal and other name changes to the post-accident monitoring system (PAMS), which interfaces with the qualified data processing system (QDPS).

- WCAP-17179, "AP1000 Component Interface Module Technical Report," Revision 2. This report discusses the design of the component interface module (CIM) within the PMS in greater detail than in the WCAP-16675-P report. The report is discussed further in Sections 7.2 and 7.9.

- WCAP-17184-P, "AP1000 Diverse Actuation System Planning and Functional Design Summary Technical Report," Revision 2. The report defines the planning process and other design attributes of the DAS. The report will be discussed in greater detail in Section 7.8.

- WCAP-17201-P, "AC160 High Speed Link Communication Compliance to DI&C-ISG-04 Staff Positions 9, 12, 13 and 15 Technical Report," Revision 0. This report discusses how the PMS will comply with certain acceptance criteria in the digital instrumentation and controls (DI&C) Interim Staff Guidance (ISG)-04. The TR will be further evaluated in Section 7.9.

- WCAP-17226-P, "Self-Powered Detector Signals in the AP1000 In-Core Instrumentation System", Revision 2. This report discusses the interactions between the nonsafety-related in-core instrumentation system (IIS) and the safety-related core exit thermocouples (CETs). As the report covers the interaction of safety-related CETs and the IIS, this report will be covered in Section 7.5.7.

- APP-GW-J0R-012, "AP1000 Protection and Safety Monitoring System (PMS) Computer Security Plan," Revision 1. This report discusses how the PMS is constructed in an SDOE and will be discussed further in Section 7.9

- APP-GW-GLR-137, "Bases of Digital Overpower and Overtemperature Delta-T ($OP\Delta T/OT\Delta T$) Reactor Trips," Revision 1. This report, also known as APP-GW-GLR-005, discusses the changes associated with how the PMS calculates its $OP\Delta T$ and $OT\Delta T$ signals. In addition to the content in Section 7.2, Appendix 7A provides additional information regarding this TR.

The staff reviewed WCAP-16592-P, Revision 2, and determined the information in the report adequately addresses the subject matter related to hazards or risks associated with the use of software or programmable technology in the PMS with one notable exception. Revision 1 of the report fails to discuss the potential hazards and/or risks associated with utilizing a firmware-based device, such as the CIM or safety remote node controller (SRNC) within the CIM subsystem of the PMS. While the CIM and SRNC will not use any software during their operation, the use of a programming language and other similar protocols during device development must be addressed and any hazards associated with the CIM and SRNC development process must be adequately mitigated or eliminated prior to their use within the PMS. Based upon the additional information required by the staff to make a determination of acceptability, the applicant submitted a response to request for additional information (RAI)-SRPSHA-01, Revision 1, on June 28, 2010. The staff finds the commitments to update the SHA, including the removal of text stating that since the CIM and SRNC execute no software when operating, there is no need to cover their development or operation in the SHA, are acceptable. Additionally, in the response, the applicant committed to adding several potential hazards and mitigation strategies based upon the use of the CIM system in the PMS in Section 5 of the SHA. The staff previously identified this issue as Open Item OI-SRP7.1-ICE-02. In a subsequent revision to the SHA, the applicant included the potential hazards and mitigation strategies, which resolves this issue.

The applicant replaced the remote shutdown workstation with the remote shutdown room (RSR). The staff finds that the change complies with all applicable acceptance criteria. Specifically, according to design drawings presented to the staff, the RSR transfer switch exists in a hallway located between the main control room (MCR) and the RSR. The staff asked the applicant to explain how it provides positive control of the reactor at all times, in accordance with 10 CFR 50.54(k), "Conditions of licenses." The applicant responded to RAI-SRP7.1-ICE-13 by stating that control of the RSR switch would be primarily through administrative or procedural control. In a "normal" evacuation of the MCR, licensed operators would pre-staff the RSR before transferring reactor control. In an extreme life-threatening emergency, the licensed operators would be directed to manually scram the reactor before exiting the MCR, and, during the short journey to the RSR, move the transfer switch to RSR control and execute the necessary operator actions that follow a reactor scram. The staff finds this approach acceptable.

In response to RAI-SRP7.1-ICE-14, the applicant provided additional information regarding the overall makeup of its electrical distribution system. The staff reviewed the proposed grounding system for instrumentation and computer systems. Section 8.3.1.1.7 of the AP1000 DCD describes the four primary grounding systems for the AP1000. In particular, the instrument and computer systems use a separate radial grounding system. Based upon the information in the referenced section and the diagrams in Chapter 8 for the Class 1E instrumentation power distribution system, the staff finds the grounded, three-phase wye transformer configuration acceptable for the I&C systems.

In Section 7.1.7, "References," of Revision 17 of the AP1000 DCD, the applicant deleted several references from the certified reference section without sufficient basis. As a result, the staff identified this as Open Item OI-SRP7.1-ICE-03. In its response to Open Item OI-SRP7.1-ICE-03, dated June 22, 2010, the applicant committed to update key reference documents in Chapter 7 of the AP1000 DCD, Tier 2, as well as update Tier 2, Table 1.6-1. Additionally, the applicant committed to update its Tier 2* document list with WCAP-16361-P and WCAP-17179-P, and restore WCAP-15927, Revision 2, to the Tier 2* list as well. In addition, the applicant committed to remove all references to APP-GW-GLR-104, and other associated references to cyber security, as the staff's review under 10 CFR Part 50, "Domestic licensing of production and utilization facilities," does not encompass a cyber security review. Further information on cyber security is provided in Chapter 13 of the SER for the respective COL. Based upon the staff's review of the applicant's commitments in the Open Item OI-SRP7.1-ICE-03 response, the staff finds the changes are acceptable. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff could not identify several of the abbreviations the applicant used in some of its originally amended TRs. In RAI-SRP7.1-ICE-25, the staff asked the applicant to define all abbreviations in the text and diagrams in TR-89, and similar documents, as well as terms such as "hardwired," "bypass function," "partial trip mode," and "failsafe trip." The NRC discussed these issues with the applicant's technical staff on January 29-30, 2009, at its Rockville, Maryland office. The staff received adequate responses to RAI-SRP7.1-ICE-24, RAI-SRP7.1-ICE-25, RAI-SRP7.2-ICE-02, RAI-SRP7.2-ICE-03, and RAI-SRP7.2-ICE-06; which the applicant has incorporated into the current revision of WCAP-16675-P and other affected TRs.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

7.1.6 Defense-in-Depth and Diversity Assessment of the AP1000 Protection System

Section 7.8 of this report discusses the safety evaluation of changes to the approved PRA-based diversity and defense-in-depth (D3) design.

7.2 Reactor Trip System

7.2.2 Protection and Safety Monitoring System Description

As described in Revision 17 of the AP1000 DCD, the applicant identified the Common Q platform as its safety-related protection system platform, negating the reference to WCAP-13383, "AP600 Instrumentation and Control Hardware and Software Design Verification and Validation Process Report," since that report primarily dealt with the use of the Eagle 21 platform, which will not be used in the AP1000 design. Specifically, in Section 7.1 of the AP1000 DCD Tier 2, the applicant removed references to the use of the Eagle 21 protection system hardware and committed to use the Common Q platform. This change was verified in the applicant's response to Open Item OI-SRP7.1-ICE-03, dated June 22, 2010, which the staff finds acceptable. As a result, removal of WCAP-13883 from the AP1000 DCD, Tier 2, Chapter 7, is acceptable.

In WCAP-16675-P, Revision 5, the applicant provided design information on the AP1000 PMS. Additionally, the applicant submitted WCAP-16674-P, Revision 4, which addressed data communications and manual component control of safety-related systems, and WCAP-17179-P, Revision 2, which discussed a subsystem within the PMS responsible for allowing nonsafety-related control of safety-related components during normal plant operation known as the CIM system. In the event of a PMS actuation, the PMS overrides nonsafety-related control through the CIM and takes the safety-related components to their safe state. This task is accomplished through priority logic within the CIM. The CIM system resides within the engineering safety features actuation system (ESFAS) portion of the PMS. The CIM system's primary active components are the CIM and SRNC.

The staff reviewed the design and architecture of the PMS to assess how it meets regulatory requirements and addresses the acceptance criteria associated with an I&C safety system. The PMS encompasses the functions of the reactor trip system (RTS), ESFAS, PAMS, and QDPS. Section 7.5 of this report includes additional information regarding the QDPS.

The discussion of the PMS in the following sections includes information presented in the publicly available, non-proprietary version of the following reports:

- APP-GW-GLR-137, Revision 1 (also known as APP-GW-GLR-005)
- WCAP-16674-NP, "AP1000TM I&C Data Communication and Manual Control of Safety Systems and Components," Revision 4
- WCAP-16675-NP, "AP IOOOTM Protection and Safety Monitoring System Architecture Technical Report," Revision 5
- WCAP-17179-NP, Revision 2
- WCAP-16438-NP, APP-GW-JJ-002-NP, "FMEA of AP 1000 Protection and Safety Monitoring System," Revision 2
- WCAP-16592-NP, Revision 2

7.2.2.1 PMS Functional Requirements

The PMS performs the functions of the RTS, ESFAS, PAMS, and QDPS. During normal operation, administrative procedures and plant control systems maintain the reactor in a safe state, preventing damage to the three barriers (fuel clad, reactor coolant system (RCS), and reactor containment building) that prevent the spread of radioactive material to the environment. Accident conditions causing one or more of the barriers to be threatened could occur. Thus, the PMS monitors key plant parameters and automatically initiates various protective functions to prevent the violation of any of the three barriers. When violation of a barrier cannot be prevented, PMS will attempt to maintain the integrity of the remaining barriers. This ensures that, given a design-basis event, the site boundary radiation releases will be maintained below the limits in 10 CFR Part 100, "Reactor site criteria." The system functions by actuating a variety of equipment and by monitoring the plant process using a variety of sensors and operations that perform calculations, comparisons, and logic functions, based on those sensor inputs. The PMS functional requirements documents discuss the protective functions of the system and the requirements these functions place on the equipment that performs them.

7.2.2.1.1 Reactor Trip Functions

The PMS generates an automatic reactor trip for the following conditions:

- source range high neutron flux trip
- intermediate range high neutron flux trip
- power range high neutron flux trip (low setpoint)
- power range high neutron flux trip (high setpoint)
- power range high positive flux rate reactor trip
- overtemperature delta-T ($OT\Delta T$) reactor trip
- overpower delta-T ($OP\Delta T$) reactor trip
- reactor trip on low pressurizer pressure
- reactor trip on low reactor coolant flow
- reactor trip on reactor coolant pump (RCP) underspeed
- RCP bearing water temperature
- pressurizer high-pressure reactor trip
- pressurizer high-water-level reactor trip
- reactor trip on low-water level in any steam generator
- high-2 steam generator water level in any steam generator
- automatic depressurization systems (ADSs) actuation reactor trip
- core makeup tank (CMT) actuation reactor trip
- reactor trip on safeguards actuation
- manual reactor trip

Revision 15 of the AP1000 DCD identified the reactor trip functions listed above. The applicant modified the design description of several reactor trip functions in Revisions 16 and 17 of the AP1000 DCD. Specifically, the applicant proposed the following modifications:

- The source range high neutron flux trip is delayed when the detector's high-voltage power supply is energized to prevent a spurious trip. The staff finds this acceptable, since the detector would be energized before being declared operable.

- The equations for performing the OTΔT and OPΔT reactor trips were modified. The staff determined, through an examination of TR, APP-GW-GLR-137, Revision 1, and associated RAIs and satisfactory applicant responses, that the new equations were equivalent to the previous equations but in a different format. Therefore, the changes are acceptable. Appendix 7A provides a more detailed evaluation of APP-GW-GLR-137. The revision of the WCAP-16361-P report will be addressed in Section 7.2.7 of this report.

- Reference to Permissive P-8 (power range nuclear power above setpoint) was removed, as Permissive P-10 made it redundant. The staff finds the change acceptable.

- Reactor Trip on High RCP Bearing Water Temperature. The trip has been modified to occur if any RCP experiences a high bearing water temperature without the P-10 interlock being able to block the trip when reactor power is below the P-10 setpoint. As this modification is conservative in nature, in that the number of permissives needing to be satisfied before the trip occurs has been reduced, the change is deemed acceptable.

- The applicant removed the automatic rod withdrawal block, which prevents rod withdrawal if the negative flux rate setpoint is exceeded, in Revision 16 of the AP1000 DCD. As a result, based upon the removal of the associated text and table information in Chapter 7 of Revision 17 of the AP1000 DCD, the staff understands the previously discussed P-17 interlock to have been removed from use in the PMS. Since this control action utilized the nonsafety-related rod control system for its actuation or, in the case of the block signal, the lack of actuation, the removal of the block action is beyond design basis as it pertains to this evaluation and is, therefore, acceptable.

The NRC determined that the other changes to the reactor trip functions, such as editorial changes or changes that added conservatism to the design, were minor; therefore, this chapter does not discuss them further.

7.2.2.1.2 Engineered Safety Features Actuation System Functions

The PMS performs both reactor trip and ESFAS functions. The AP1000 design provides I&C to sense accident situations and initiate engineered safety features (ESFs). The occurrence of a limiting fault, such as a loss-of-coolant accident (LOCA) or a secondary system break, requires a reactor trip plus actuation of one or more of the ESFs. This combination of events prevents or mitigates damage to the core and RCS components and provides containment integrity.

The PMS is actuated when safety system setpoints are reached for selected plant parameters. The selected combination of process parameter setpoint violations is indicative of primary or secondary system boundary challenges. Once the system receives the required logic combination, the PMS equipment sends the signals to actuate appropriate ESF components.

The PMS initiates the following ESF system-level actuations:

- safeguards actuation
- containment isolation
- in-containment refueling water storage tank (IRWST) injection
- CMT injection
- ADS

- RCP trip
- main feedwater isolation
- passive residual heat removal actuation
- turbine trip
- containment recirculation
- steamline isolation
- steam generator blowdown system isolation
- passive containment cooling actuation
- startup feedwater isolation
- boron dilution block
- chemical and volume control system (CVS) isolation
- steam dump control
- MCR isolation
- auxiliary spray and purification line isolation
- containment air filtration isolation
- refueling cavity isolation
- CVS letdown isolation
- pressurizer heater block
- steam generator relief isolation
- normal residual heat removal containment isolation
- demineralized-water transfer and storage system isolation
- reactor vessel head vent valve control

Revision 15 of the AP1000 DCD identified the ESF system-level actuations listed above. The applicant incorporated several minor changes related to the ESF functions into Section 7.3 of the AP1000 DCD, and the staff finds the changes acceptable.

7.2.2.1.3 Component Control Functions

The applicant provides control of individual safety-related components that perform Class 1E functions. Component-level control consists of the following functions:

- resolution of multiple demands (priority logic) for a given component from various systems
- application of manual component demands
- performance of the component protection logic (e.g., torque limit, antipump latch)
- reporting of component status to the plant information system
- local component control

The following inputs are required for control of individual components:

- automatic system-level actuation commands from the reactor trip and ESF actuation logic

- manual system-level actuation commands from the fixed position switches in the MCR and RSR

- individual safety component control commands from the nonsafety-related plant control system (PLS) for component actuations with no onerous consequences (for test, maintenance, restoration, and noncredited actuations)

- individual safety component control commands from the safety and QDPS displays in the MCR for component actuations with onerous consequences

- component feedback signals from the individual safety components to the PMS

The outputs to individual safety-related components consist of hardwired control signals to open or close a solenoid valve, motor-operated valve (MOV), or circuit breaker.

7.2.2.2 AP1000 Protection and Safety Monitoring System Operation

The PMS detects off-nominal conditions and actuates appropriate safety-related functions necessary to achieve and maintain the plant in a safe-shutdown condition. The PMS controls safety-related components in the plant that are operated from the MCR or RSR workstation. In addition, the PMS provides the equipment necessary in its QDPS subsystem to monitor the plant's safety-related functions during and following an accident, as identified in regulatory guide (RG) 1.97, "Criteria for Accident Monitoring Instrumentation for Nuclear Power Plants," Revision 3.

The AP1000 PMS consists of four redundant divisions, designated A, B, C, and D. The PMS performs the necessary safety-related signal acquisition, calculations, setpoint comparison, coincidence logic, reactor trip and ESF actuation functions, and component control functions to achieve and maintain the plant in a safe shutdown condition. The PMS also contains maintenance and test functions to verify proper operation of the system. The PMS includes four redundant safety displays, one for each division and two QDPS displays, located on the primary dedicated safety panel (PDSP) in the MCR. Only two of the four divisions contain software to drive QDPS displays and to provide PAMS information to the operator, on Divisions B and C. The system's use of four redundant divisions is one of the mechanisms employed to satisfy single-failure criteria and improve system availability.

The I&C equipment performing reactor trip and ESF actuation functions, its related sensors, and the reactor trip switchgear are, for the most part, four-way redundant. This redundancy permits the use of bypass logic, so that a division or an individual channel within a division taken out of service can be accommodated by the operating portions of the protection system that revert to a two-out-of-three (2oo3) logic function from a two-out-of-four (2oo4) logic function. Additional discussion related to bypasses and the indication of inoperable channels within the PMS can be found in Section 7.5.6 of this report.

Four redundant measurements, using four separate sensors, are made for each variable used for reactor trip or ESF actuation within a single division. Each division, which is comprised of two redundant channels, processes one measurement. Analog signals are converted from remote sensor outputs to digital form by analog-to-digital converters within the division's bistable processor logic (BPL) modules - one per channel. Signal conditioning is applied to selected inputs to the BPL, after the digital conversion. Following necessary calculations and

processing, the measurements are compared against the applicable setpoint for that variable. A partial trip signal for a parameter is generated in the BPL if the channel's measurement exceeds its predetermined or calculated limit. Processing of variables for a reactor trip is identical in each of the four redundant divisions of the protection system.

Two local coincidence logic (LCL) processors within each division receive the output signal from each division's BPLs. The LCL subsystem acts to initiate a reactor trip or ESF actuation when a predetermined condition in 2oo4 independent safety divisions reaches a partial trip or partial actuation state. The LCL also provides for the bypass of trip or actuation functions to accommodate periodic tests and maintenance.

7.2.2.2.1 Reactor Trip Operation

The reactor trip coincidence logic performs the logic function to combine the partial trip signals from the BPL subsystems and generates a failsafe trip output signal to the reactor trip switchgear. The reactor trip signal from each of the four divisions of the PMS is sent to the division's reactor trip circuit breakers (RTCBs). Each division controls two RTCBs. The design of the RTCB placement ensures that the reactor trip would still occur, even in the presence of a failure of one of the division's two RTCBs. The reactor is tripped when two or more divisions actuate, thereby generating a reactor trip signal opening their breakers. The automatic trip demand signal initiates the following two actions: it deenergizes the undervoltage trip coils on the RTCBs, and it energizes the shunt trip devices on the RTCBs. Either action causes the breakers to trip. Opening the appropriate trip breakers by any two divisions removes power to the rod drive mechanism coils, allowing the rods to fall into the core. This rapid negative reactivity insertion causes the reactor to shut down. Section 7.2.2.3.7, "Capability for Test and Calibration," provides further information on testing RTCBs.

7.2.2.2.2 ESF Actuation

The ESF coincidence logic processors perform the logic function to combine the partial actuation signals from the BPL subsystems, along with automatic and manual permissives, blocks, and resets, to generate a fault tolerant actuation output signal to the integrated logic processor (ILP) subsystems.

The primary functions of the ESF logic processors are to process inputs, calculate system-level actuation, combine the automatic actuation with the manual actuation and manual bypass data, and transmit the data to the ILPs. To perform the ESF coincidence logic calculations, the ESF processors require data from the BPL subsystems and also use manual inputs (such as setpoints and system-level blocks and resets) from the MCR and the RSR workstation.

The ESF logic processors perform the following functions:

- Receive bistable data supplied by the four divisions of BPL subsystems and perform 2oo4 voting on this data.
- Implement system-level logic and transmit the output to the ILP processors for ESF component fan-out and actuation.
- Process manual system-level actuation commands received from the MCR and/or RSR.

The ESF component control function uses redundant ILPs, which serve as LCL output control signal "fan-out" devices, which distribute the activate signal to the various SRNCs that forward their given output to their respective CIMs. The CIMs provide a distributed interface between the safety system and plant equipment for control of non-modulating safety-related plant components. The safety-related input to the CIM comes from both of its respective SRNCs within the PMS, while the nonsafety-related input enters the CIM through the nonsafety-related remote node controller (RNC). Non-modulating control relates to the opening or closing of solenoid valves and solenoid pilot valves and the opening or closing of MOVs and dampers. It also provides the plant operator with information on the equipment status, such as an indication of component position (full closed, full open, valve moving), component control modes (manual, automatic, local, remote), or abnormal operating conditions (power not available, failure detected).

The staff approved the Common Q platform portion of the AP1000 PMS, previously based on information in the Common Q topical report. However, regarding the use of the ESFAS functions, the PMS design uses distribution devices known as ILPs, SRNCs, and CIMs, which had not been previously evaluated in the Common Q platform. The applicant provided design information detailing the functions of each of the aforementioned components. Sections 7.2.2.3.13 through 7.2.2.3.15 include the technical discussion regarding the additional information the staff reviewed for each of these components before determining their acceptability. The ILP is an intra-divisional device that receives its input from the LCL and distributes its outputs to a given SRNCs which forward their outputs to their assigned CIM. Besides distributing the activate signals to non-modulating safety-related devices, the CIM serves as an interface control device, as it receives inputs from both the safety-related PMS and the nonsafety-related PLS to actuate the requested vital components of the safety-related ESFAS.

7.2.2.3 PMS Evaluation

The staff evaluated the technical requirements of the PMS design against the requirements and acceptance criteria in the following documents:

- 10 CFR 50.49, "Environmental qualification of electric equipment important to safety for nuclear power plants"

- 10 CFR 50.55a(h), "Codes and standards," which incorporates by reference Institute of Electrical and Electronic Engineers (IEEE) Standard (Std.) 603-1991, "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations" (NOTE: The clauses of IEEE Std. 603-1991 not referenced in the topical-based discussion below are unaffected by the proposed changes described in Revision 19 of the AP1000 DCD).

- 10 CFR Part 50, Appendix A, "General Design Criteria [GDC] for Nuclear Power Plants"

  - GDC 13, "Instrumentation and Control"
  - GDC 20, "Protection System Functions"
  - GDC 21, "Protection System Reliability and Testability"
  - GDC 22, "Protection System Independence"
  - GDC 23, "Protection System Failure Modes"
  - GDC 24, "Separation of Protection and Control Systems"
  - GDC 29, "Protection Against Anticipated Operational Occurrences"

The industry and staff guidance in Chapter 7 of NUREG-0800 applies to the PMS review.

The staff evaluated the PMS architecture against those requirements affected by the additional or modified design information provided on the PMS. Those requirements include single-failure protection, quality, equipment qualification, system integrity, independence, capability for test and calibration, information displays, control of access, repair, automatic and manual control, and bypasses. Section 7.2.7 discusses setpoints. The staff determined that other requirements for I&C systems were not affected by changes in Revision 19 of the AP1000 DCD, compared to those approved in Revision 15. Specifically, the applicant uses the Common Q platform as a major portion of the PMS. This report evaluated major subcomponents of the PMS, including those that were not previously developed at the time of the staff's original SER for the AP1000 such as the ILP, SRNC, and CIM. Sections 7.2.2.3.13 through 7.2.2.3.15 discuss those evaluations.

7.2.2.3.1 Common Q Evaluation

The staff previously approved the use of the Common Q platform for generic nuclear power plant (NPP) applications. Section 7.2.2.2.2 of this report includes reference material related to the Common Q topical report and the associated safety evaluations. The applicant presented additional planning and design information related to the use of the Common Q platform in the AP1000 design in the form of the newly submitted TRs related to the PMS that are discussed in Section 7.1.5.

7.2.2.3.2 Single-Failure Protection

The staff evaluated the single-failure protection characteristics of the Common Q and PMS against requirements in IEEE Std. 603-1991, Clause 5.1, "Single Failure Criterion," and GDC 21. The staff used the guidance in RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems," which endorses IEEE Std. 379-1988, "Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station Safety Systems."

To address the guidance in RG 1.53, the applicant presented the NRC with an FMEA and an SHA to demonstrate the system's capability to withstand a single-failure event. WCAP-16438-P, Revision 2, describes the FMEA, and WCAP-16592-P, Revision 2 describes the SHA for the PMS. The staff's evaluation of the PMS SHA is described in Section 7.1.5, "Instrumentation and Control System Architecture."

The staff examined the AP1000 FMEA against the guidance of IEEE Std. 352-1998, "IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Safety Systems," and IEEE Std. 379-1988. The staff finds the report provides a satisfactory demonstration of the system's fault tolerance under various scenarios. Several technical questions were identified regarding specific system responses in the presence of certain failure types and required resolution as shown in the following examples:

- Five CIMs possess a mean time between failure (MTBF) that did not meet the design reliability targets for the PMS. However, the text gave no explanation or analysis as to how the system will minimize, mitigate, or eliminate these failure types. To address this issue, the applicant issued a response to RAI-SRP7.1-FMEA-01 in December 2009, stating that an MTBF analysis would be conducted and was expected to be completed by July 2010. The response further discusses how, once the MTBF analysis report is

completed, if the failure rates were not in line with the expected failure rate of at least 200,000 hours, the applicant's design change process would be utilized to modify the equipment to reach the MTBF goal or adequately explain why the higher rate of failure is acceptable. The staff finds the use of an MTBF analysis to verify a sufficiently low level of failures of the CIM and SRNC to be acceptable.

- According to the FMEA, when the check sum verification process fails, the setpoints are not updated; however, this condition would be restricted to one division only. The FMEA then discusses how the division may trip erroneously or not trip due to the setpoints not being properly updated. In its response dated December 1, 2009, the applicant stated, in part, that the cyclic redundancy check (CRC) verification mechanism can fail and the correct setpoint value still be monitored by the maintenance and test panel (MTP). Although the MTP may not receive the updates due to a failure of the CRC, there are continuous lower level diagnostics that provide alarm and indication informing the operators that a fault has occurred between the LCL and the MTP. Additionally, even though the diagnostics and self-tests that are executing continuously are not credited, the fault would be limited to a single division's MTP and is, therefore, acceptable.

The FMEA describes how the PMS, through the MTP, periodically adjusts setpoints automatically. This automatic adjustment of setpoints, without operator control, would not meet the requirements in Criterion III, "Design Control," in 10 CFR Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." In its response dated November 9, 2009, the applicant related the "periodic refreshing" of data was part of the inherent communications in a deterministic format utilized between the AC160 (Common Q processor) and the MTP. The response stated further that the setpoints within the AC160 can only be altered manually by the operator of the MTP. The applicant submitted Revision 3 of the FMEA to more clearly state that the system is not continually sending "new" updated setpoints to the AC160, but simply performing its communication function. The staff finds this revision acceptable.

When comparing the requirements in Clause 5.1 of IEEE Std. 603-1991 and GDC 21 to the information in WCAP-16675-P and WCAP-16438-P, and after reviewing the responses provided by the applicant related to the FMEA, the staff determined the single failure requirements related to IEEE Std. 603-1991 have been adequately addressed. The NRC staff previously identified these issues as Open Item OI-SRP7.2-ICE-01, and as a result of the staff's review of the responses provided by the applicant and its commitment to update the next revision of the FMEA based upon their responses to RAI-SRP7.1-FMEA-02, RAI-SRP7.1-FMEA-04, RAI-SRP7.1-FMEA-05 and RAI-SRP7.1-FMEA-07 the staff finds the design satisfies the single failure criterion. In Revision 3 to the FMEA, the applicant incorporated the changes described in the RAI responses, which resolves this issue.

7.2.2.3.3 Quality

Clause 5.3 of IEEE Std. 603-1991 and 10 CFR Part 50, Appendix B, require safety-related I&C systems to be designed, manufactured, inspected, installed, and tested under an acceptable quality assurance program. NUREG-0800 Appendix 7.1-D, "Guidance for Evaluation of the Application of IEEE Std. 7-4.3.2"; Section 5.3; and BTP 7-14 specifically address the criteria for a quality software development process. Additionally, the staff evaluated the documentation in the SLC for the Common Q portion of the PMS against the guidance in the following documents:

- RG 1.168, "Verification, Validation, Reviews and Audits for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," which endorses IEEE Std. 1012-1998, "IEEE Standard for Software Verification and Validation," and IEEE Std. 1028-1997, "IEEE Standard for Software Reviews and Audits"

- RG 1.169 "Configuration Management Plans for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," which endorses IEEE Std. 828-1990, "IEEE Standard for Software Configuration Management Plans"

- RG 1.170, "Software Test Documentation for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," which endorses IEEE Std. 829-1983, "IEEE Standard For Software Test Documentation"

- RG 1.171, "Software Unit Testing for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," which endorses IEEE Std. 1008-1987, "IEEE Standard for Software Unit Testing"

- RG 1.172, "Software Requirements Specifications for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," which endorses IEEE Std. 830-1993, "IEEE Recommended Practice for Software Requirements Specifications"

- RG 1.173, "Developing Software Life Cycle Processes for Digital Computer Software Used in Safety Systems of Nuclear Power Plants," which endorses IEEE Std. 1074-1995, "IEEE Standard for Developing Software Lifecycles and Processes"

In Section 7.2.2.2, the staff discusses the approval of the Common Q platform topical report. However, additional requirements were placed on the SLC for the AP1000 safety system, beyond those described in the Software Program Manual (SPM), WCAP-16096-NP-A, "Software Program Manual for Common Q Systems," Revision 1A, dated January 21, 2005. WCAP-15927, "Design Process for AP1000 Common Q Safety Systems," Revision 0, placed additional requirements on the design and testing teams developing the PMS. The certified design (Revision 15) of the AP1000 DCD designated the SPM and WCAP-15927 as Tier 2* documents, requiring NRC approval before altering the applicant's commitments. The staff recently received, reviewed, and approved the use of WCAP-15927, Revision 2. Section 7.2.5 provides further information concerning this matter.

In AP1000 DCD Tier 1, Table 2.5.2-8, Design Description 11, the applicant proposed the removal of design requirements and system definition phases of the PMS SLC ITAAC. The design requirements phase corresponds to the planning activities phase of the SLC in NUREG-0800 BTP 7-14, and the system definition phase corresponds to the requirements activities phase in the same NUREG-0800 BTP. The applicant made available for audit the software planning documents to support removal of the design requirements phase. The staff audited those software planning documents on several occasions, which are discussed below. The applicant describes its quality software development process for the AP1000 project in the Common Q SPM and WCAP-15927, Revision 2, documents. The staff reviewed the proprietary and nonproprietary documentation associated with the first two phases of the SLC as it relates to Common Q portion of PMS system development. Further information regarding lifecycle development and completion as it relates to the PMS is provided in Section 7.2.5.

The applicant originally provided 11 proprietary documents that comprise the design requirements phase of the AP1000 SLC. On April 9-11, 2008; October 9-16, 2008; January 22-30, 2009; and July 30, 2009, the staff conducted site visits at the Westinghouse Twinbrook location in Rockville, Maryland, to review the proprietary documents associated with the design requirements phase. In RAI-SRP7.1-ICE-03, the NRC asked the applicant to explain, in the Tier 2 information of the DCD, how it meets the requirements of the planning ITAAC. This includes a diagram of the planning process and sufficient planning documentation related to the lifecycle development plan for the Common Q and CIM system within the PMS, and the steps taken by the applicant to ensure the PMS is constructed in a SDOE and other project-specific documents. The lack of an adequately detailed CIM system programmable technology lifecycle was captured under Open Item OI-SRP7.2-ICE-05 and the discussion of how the applicant intends to ensure the PMS is constructed in a SDOE will be covered in Section 7.9. The issue of which documents delineate the design requirements or planning activities phases of the PMS developmental lifecycle and their relationship to one another was reconciled in Section 7.2.5 prior to the staff's acceptance of the design requirements phase of the PMS lifecycle development process as complete. As a result, the issue previously identified as Open Item OI-SRP7.2-ICE-02 is considered complete based upon the commitments made in the revised response to Open Item OI-SRP7.2-ICE-02 to include an adequately detailed discussion of the design requirements in Chapter 7 of the DCD. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Several proprietary documents listed as proof of completion of the design requirements phase detail the relationship between two Westinghouse organizations: Repair, Replacement, and Automation Services (RRAS) and Nuclear Power Plants (NPP) (e.g., RRAS AP1000 NuStart I&C Program Project Plan (WNA-PN-00031-GEN[3]) and RRAS AP1000 NuStart I&C Program Project Quality Plan (WNA-PQ-00166-GEN[4])). The documents reveal how the subcontractor (Westinghouse RRAS) interfaces with the parent organization (Westinghouse NPP); however, they originally did not describe how Westinghouse NPP interfaces with, and holds accountable, Westinghouse RRAS, employees, and other subcontractors. The NRC requested this information in RAI-SRP7.1-ICE-04. The applicant provided a written RAI response stating that its RRAS organization will complete all work on safety systems (i.e., all work performed on the I&C safety systems for the AP1000 will be conducted by either RRAS or its subcontractors, who will be required to function under Revision 5 of the Westinghouse Quality Management System (QMS) Manual). The NRC accepted the applicant's use of Revision 5 of the QMS Manual to satisfy the requirements in Appendix B to 10 CFR Part 50. This satisfies RAI-SRP7.1-ICE-04. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In July 2009, the staff conducted an audit of the recently developed proprietary PMS Test Plan, the SRNC and CIM Project Plan, and the CIM System Test Plan documents to ensure compliance with requirements and the applicant's commitments. Overall, the NRC finds the results of the audit to be acceptable. Originally requested under RAI-SRP7.1-ICE-05, the documents listed above satisfy the need for sufficient test plan information related to the planning phase of the Common Q SLC; therefore, the NRC staff considers RAI-SRP7.1-ICE-05 closed.

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[3] Westinghouse altered the numbering format for this document. It is now numbered WNA-PN-00043-WAPP.

[4] Westinghouse altered the numbering format for this document. It is now numbered WNA-PQ-00201-WAPP.

Concerning the proprietary AP1000 NuStart Protection and Safety Monitoring System Software Project Plan (WNA-PJ-00071-GEN), the staff raised issues that included how the applicant could give a value of zero to the overall risk for software development when the risk in some specific areas was rated as high. Furthermore, Appendix A of the same report states that "Test/System Integration Phase Independent Verification and Validation [IV&V] is not within the scope of the AP1000 NuStart Project." The staff requested additional information in this area in RAI-SRP7.1-ICE-08, which discussed the revision to the table. At a meeting in January 2009, the applicant committed to revise notes in the Software Project Plan[5] document, WNA-PJ-00071-GEN, to inform the reader that "zero," in the case of risk, actually means minimal, not "no risk." During the staff visit to the Twinbrook facility on July 30, 2009, the staff confirmed that the applicant had incorporated this item into the proprietary software document. Therefore, this item adequately addresses the RAI, which is now considered closed.

The staff reviewed the proprietary document, WNA-PJ-00071-GEN, "AP1000 NuStart PMS Software Project Plan," and its descendent document, the AP1000/NuStart/DOE Design Finalization and Safety Monitoring System Software Development Plan, WNA-PN-00042-WAPP, as part of the design requirements phase review. Appendix A, Item 4, of WNA-PJ-00071-GEN states that the test/system integration phase IV&V is not within the scope of the AP1000 NuStart project. The software project will be frozen at the processor module software test for Division B software. The staff asked the applicant to clarify these statements; specifically, how the applicant would test Divisions A, C, and D software and the complete software of each division, beyond the processor module software tests. In a letter dated September 3, 2008, "AP1000 Response to Request for Additional Information (SRP7)," the applicant clarified the statements in WNA-PJ-00071-GEN regarding processor module software testing. The letter stated the following:

> The deliverable for the cited plan is a detailed design up to and including the software design. This includes a software demonstration system using a test bed configured for Division B. It was not intended to validate all software for all divisions. The demonstration software for Division B marks the conclusion of the plan's scope.
>
> Once the demonstration software for Division B is complete, a software development plan will be developed to complete the software development life cycle taking credit for work completed in first plan. This would include activities for unit testing, code review, channel integration test for all four divisions, and system integration test, as well as the life cycle V&V activities.
>
> The completed Division B software is a sufficient sample to close the DAC software open issue and the V&V issue, because the completed development of Division B software will demonstrate all of the representative software subroutines included in all of the divisions (i.e., the RPS, ESFAS and QDPS functions).

The staff understands the applicant's explanation that the portion of module testing to be combined and constructed into a test bed mimicking Division B of the PMS in its entirety will

[5] Westinghouse altered the numbering format for this document. It is now numbered WNA-PD-00042-WAPP. Westinghouse also retitled it, "AP1000 NuStart/DOE Design Finalization Protection and Safety Monitoring System Software Development Plan."

serve as a "proof of concept" test for PMS and QDPS software and hardware, rather than an interdivisional integration test of the PMS and QDPS, to which the applicant is committed, later in the SLC process. The staff accepts this approach and closes RAI-SRP7.1-ICE-09.

Section 7.2.5, "PMS Design Process Review," includes a detailed discussion of the level of quality the applicant requires to implement its SLC and programmable technology lifecycle for both the Common Q and CIM subsystem portions of the PMS, and provides the staff's findings as they relate to overall PMS quality.

7.2.2.3.4 Equipment Qualification

Clause 5.4, "Equipment Qualification," of IEEE Std. 603-1991 states, in part, that safety system equipment shall be qualified by type test, previous operating experience, or analyses, or any combination of these three methods, to substantiate that it will be capable of meeting, on a continuing basis, the performance requirements as specified in the design basis. The staff reviewed docketed and Westinghouse proprietary documents relating to the overall methodology and approach when dealing with equipment qualification. The Common Q topical report describes the generic equipment qualification of the Common Q equipment. Additionally, AP1000 DCD Tier 1, Table 2.5.2-8, Items 2, 3, and 4, identify ITAAC to address the seismic, electromagnetic/radio-frequency/electrostatic discharge, and temperature/humidity qualification of PMS equipment. To complete the ITAAC, a licensee incorporating the AP1000 certified design would need to verify that the generic equipment qualification for the Common Q as well as the CIM system portion of the PMS equipment includes the site-specific seismic, electromagnetic/radio-frequency/electrostatic discharge, and temperature/humidity profile. Based on the generic equipment qualification of the Common Q equipment and the ITAAC in Tier 1 of the AP1000 DCD, the staff finds the equipment qualification acceptable for the proposed DC amendment as it relates to the PMS.

7.2.2.3.5 System Integrity

Clause 5.5 of IEEE Std. 603-1991 states, in part, that safety systems shall be designed to accomplish their safety functions under the full range of applicable conditions enumerated in the design basis. The staff used the guidance in Appendix 7.1-C and Appendix 7.1-D of NUREG-0800 to review the PMS and its means of meeting the standard in Clause 5.5.

A special concern with any safety-related digital system is confirmation of its ability, in real time, to ensure completion of the protective actions within critical points of time identified in the design basis. Section 7.9 discusses real-time performance with respect to data communications. Items 6a, 6b, and 6c of AP1000 DCD Tier 1, Table 2.5.2-8, would, through their ITAAC, address the overall verification of the system response time. With regard to failure modes, the staff confirmed that the PMS design is such that reactor trip functions fail to the tripped state and ESF functions fail to the as-is state. Additionally, the Common Q equipment contains self-diagnostics to alarm failed conditions and place the system into a fail-safe state. However, the applicant does not propose to use self-diagnostic tests for technical specifications surveillance tests. Based on the ITAAC, FMEA, and self-testing features, the staff finds the PMS design acceptable from a system integrity standpoint.

7.2.2.3.6 Independence

Clause 5.6, "Independence," of IEEE Std. 603-1991 and GDC 22 and GDC 24 require, in part, that safety I&C systems be designed with physical, electrical, functional and communications

independence among redundant divisions and between safety and nonsafety systems. The staff compared the independence requirements of the Common Q system to the guidance in NUREG-0800 Appendix 7.1-C and Appendix 7.1-D. Additionally, the staff compared the independence requirements of the PMS as they relate to ensuring the protection system was developed in a SDOE. The evaluation of the system being constructed and operated in a SDOE is discussed in Section 7.9.

Based upon information provided in the Common Q topical report and its appendices, the NRC approved the Common Q platform, which includes the BPLs, the LCLs, the interface and test processor (ITP), and the MTP. While the physical and electrical independence aspects of the Common Q have been demonstrated, several issues remained. One issue concerns the inter-and intra-divisional data communications independence requirements of the Common Q platform. Section 7.9 discusses this issue.

As previously stated, within the layout of the PMS, yet external to the previously approved Common Q platform, the NRC evaluated the RNC, the SRNC, and the CIM against the independence requirements for physical separation and electrical, functional and communications independence (covered in Section 7.9) in Clause 5.6 of IEEE Std. 603-1991. GDC 13, 20, 21, 22, 23, and 24 include other requirements affecting CIM independence.

IEEE Std. 603-1991, which the NRC endorsed in 10 CFR 50.55a(h), defines "associated circuits" as non-Class 1E circuits that are not physically separated or are not electrically isolated by acceptable separation distance, safety class structures, barriers, or isolation devices.

IEEE Std. 384-1981, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," defines the acceptance criteria for the associated circuits. The RNC, which serves as an interface between the nonsafety-related PLS Ovation® platform and the safety-related CIM, serve as part of the communications isolation function, according to Section 5.2.2 of WCAP-16674-P, Revision 4. This information also appears in Section 3.3.5 of WCAP-16675-P, Revision 5.

According to the description in the TRs, electrical and physical isolation between the RNC and the safety-related CIM is achieved by the use of fiber-optic cabling between the RNC and the remote input/output (I/O) bus, which connects to the PLS side of the RNC. The applicant committed to ensuring that the RNC will undergo qualification testing that meets or exceeds the guidance in IEEE Std. 384-1981. Thus, the device will undergo the qualification requirements placed on Class 1E circuits to ensure that the Class 1E circuits are not degraded below an acceptable level, even in the presence of a failure of the associated circuit. The NRC acknowledges that the RNC will be qualified as an associated circuit, since the fiber-optic cable providing electrical isolation is between the nonsafety-related PLS and the RNC.

The staff understands the subcomponent performing communications conversion is located physically on the CIM and that it creates the functional isolation between the RNC and CIM by performing the communications protocol or language conversion from the Emerson Ovation® programming language to discrete digital signals that will be provided to the priority logic, also located on the CIM. This type of arrangement also creates the functional isolation required to satisfy NRC requirements by ensuring that a failure, degradation, or corruption of the information being received by the CIM will not disable the ability of the PMS to execute or complete its function through the priority module, in accordance with IEEE Std. 603-1991, Clauses 5.2 and 5.6.1. Specifically, the priority logic of the CIM provides the PMS priority over the nonsafety-related control system if a safety-related plant component needs to actuate to its

safe state in all cases. This issue was addressed adequately in the applicant's response to RAI-SRP7.0-ICE-03 providing additional information regarding how the PMS has priority over the PLS, even when failure conditions are present. These changes have been incorporated into WCAP-17179-P, Revision 2.

Section 7.9 includes the staff's full evaluation, covering nonsafety-related to safety-related data communications of the PMS, via the RNC and CIM, and other I&C systems.

7.2.2.3.7 Capability for Test and Calibration

The staff evaluated the Common Q system against the requirements of IEEE Std. 603-1991, Clause 5.7, "Capability for Test and Calibration," and the guidance in the following documents:

- NUREG-0800, BTP 7-17, "Guidance on Self-Test and Surveillance Test Provisions"
- NUREG-0800 Appendix 7.1-D, "Guidance for Evaluation of the Application of IEEE Std. 7-4.3.2"
- RG 1.22, "Periodic Testing of Protection System Actuation Functions," issued 1972
- RG 1.118, "Periodic Testing of Electric Power and Protection Systems," issued 1995, which endorses IEEE Std. 338-1987, "IEEE Standard Criteria for the Periodic Surveillance Testing of Nuclear Power Generating Station Safety Systems"

The staff found that the applicant meets the physical requirements that describe how testing and calibration will be performed administratively, through appropriate procedural guidance for both operational and maintenance personnel, as well as by the design of the Common Q platform and CIM subsystem, which provide various interlocks within the Common Q portion and CIM subsystem within the PMS. Based upon a review of all information in the DCD and referenced TRs, the applicant does not credit any self-testing features with the PMS as replacements for periodic technical specification surveillance tests. The staff finds the applicant's approach acceptable. The staff will continue to verify this finding through the ITAAC process in later development stages (implementation, integration, and factory acceptance testing) of the developmental lifecycles for the Common Q platform and CIM subsystem within the PMS as a whole to ensure the actual system meets the requirements described in the planning activities and requirements activities developmental stages.

The applicant stated that each individual RTCB would be opened during a trip actuation device operational test once per year. In current licensed plants, the maximum length of time between openings of RTCBs (or equivalent) is typically 92 days. This issue is discussed in Chapter 16 of NUREG-1793 and this report. This issue was determined to be adequately addressed in Revision 15 of the DCD and no changes were made to RTCB layout or design, including the periodicity of RTCB testing. Therefore, Open Item OI-SRP7.2-ICE-03 is considered resolved.

7.2.2.3.8 Information Displays

Clause 5.8 of IEEE Std. 603-1991 requires information displays for manually controlled actions, safety-system status, and indication of bypasses. Additionally, the information displays should be accessible to operators. The staff used the guidance in RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems," issued 1973, during the review.

The information displays used by the PMS remain unchanged since they were found to be acceptable in the "Closeout of Category 1 Open Items for the Common Q Systems," with the caveat that the flat panel display systems (FPDS) be qualified. Specifically, the NRC's SER, "Safety Evaluation for the Closeout of Several of the Common Qualified Platform Category 1 Open Items Related to Reports CENPD-396-P, Revision 1, and CE-CES-195, Revision 1" (TAC Number MB0780), dated June 22, 2001, includes the following statement:

> PSAI 6.3 has been resolved generically and may be closed as a requirement in the SE. In connection with its resolution, the staff also found that the FPDS, subject to its successful performance during scheduled qualification testing, to be acceptable for use in Common Q designs as a Class 1E HMI device.

Therefore, a licensee incorporating by reference the AP1000 certified design would need to specifically address the equipment qualification of the FPDS when closing out ITAAC identified in AP1000 DCD Tier 1, Table 2.5.2-8, Design Commitments 2, 3, and 4. The staff finds the AP1000 information display design acceptable as it relates to Clause 5.8 of IEEE Std. 603-1991. Section 7.5 includes further discussion of information displays important to safety. Chapter 18, "Humans Factors Engineering," evaluates the adequacy of display information presented to operators.

7.2.2.3.9 Control of Access

Clause 5.9, "Control of Access," of IEEE 603-1991 requires control of access to the PMS. The staff judged the physical capability of the PMS to provide for access control to be acceptable with regard to its system design and administrative controls prior to the removal of the planning activities and requirements activities ITAAC. Section 7.9 includes the evaluation of the ability of the Common Q platform and CIM subsystem within the PMS to meet these requirements to ensure that the PMS is planned and constructed within a SDOE and a protected data communications standpoint. Additionally, Section 7.2.7 discusses the access controls to restrict the alteration of setpoints.

7.2.2.3.10 Repair

The staff compared the information the applicant presented in WCAP-16675-P regarding the capability of the PMS to undergo repair activities both online and offline to meet the requirements of IEEE Std. 603-1991; Clause 5.10, "Repair"; and adequately address the guidance in BTP 7-17. The staff finds that although online self-diagnostics may aid the operator, maintenance, or engineering team to diagnose a fault condition, they do not replace the need for surveillance testing as discussed in Section 7.2.2.3.7.

7.2.2.3.11 Automatic and Manual Control

Clauses 6.1 and 7.1 of IEEE Std. 603-1991 require provisions for automatic control of safety functions. Specifically, Clause 6.1 requires, in part, that means be provided to automatically

initiate and control all protective actions except as justified in Clause 4.5. The safety-system design shall be such that the operator is not required to take any action before the time and plant conditions specified in Clause 4.5. Clause 7.1 requires the execute features to receive and act upon automatic controls.

Sections 1.1 and 1.2 of WCAP-16675-P identify the automatic reactor trip and ESF actuations. The applicant did not identify any manual controls for reactor trip or ESF actuations for the PMS in place of automatic controls. Additionally, the reactor trip breakers and ESF equipment are designed to accept and act upon automatic commands from the PMS. Therefore, the staff finds that the PMS design meets Clauses 6.1 and 7.1 of IEEE Std. 603-1991.

Clauses 6.2 and 7.2 of IEEE Std. 603-1991 require provisions for manual actuation of safety functions. Specifically, Clause 6.2 requires, in part, system-level manual actuation of safety functions that minimize the number of discrete operator manipulations and depend on the operation of a minimum amount of equipment, consistent with the constraints of Clause 5.6.1, "Independence Between Redundant Portions of the Safety System." Position C.4 of RG 1.62, "Manual Initiation of Protective Actions," issued October 1973, states that equipment common to both the automatic and manual initiation should be kept to a minimum. The action-sequencing logic may be common, as long as component-level control of safety equipment is provided in the control room. Figure 2-2 of WCAP-16675-P identifies the voting logic, as well as the action-sequencing logic, common to the automatic and manual ESF initiation paths. The staff finds that, while this departs from the guidance in RG 1.62, the design is acceptable for the following reason. In the event there is a failure that would disable common portions of the automatic and manual initiation paths in all divisions, a DAS is provided that is not common to the safety-related PMS. The point at which the system-level manual actuation is brought into PMS architecture minimizes the use of equipment and, therefore, reduces the complexity associated with implementing the system-level manual actuation. From a single-failure standpoint, both the automatic and manual initiation paths are designed for single-failure protection. For the RTS-level manual actuation, the switches in the MCR are hardwired to the reactor trip breakers. The staff finds the system-level manual actuation of the reactor trip and ESF to be acceptable.

Section 3.4.2 of WCAP-16675-P describes manual component-level control. The proposed design provides safety-related component-level control of components having onerous consequences, using the soft controls on the safety displays in the MCR and from the CIMs in the equipment rooms. Onerous consequences are defined as those that cause a breach of the RCS pressure boundary or cause a need to shut down the plant to cold conditions to effect repairs. Component-level control of equipment not having onerous consequences is provided through the nonsafety-related workstations in the MCR or the RSR in the event of a condition requiring evacuation of the MCR. The staff finds the proposal to be acceptable, since there is no regulatory requirement for component-level control as it relates to the AP1000 design. Specifically, the plant design does not credit component-level control of equipment to perform safety functions; all safety functions are performed at the system level. The applicant has defined the components having onerous consequences in Table A-1 of the FMEA of AP1000 PMS, WCAP-16438-P, Revision 3, and the staff finds this list acceptable. Although the applicant considers the information proprietary, it will make the component information available to operations, engineering, and maintenance personnel in a licensed power plant using the AP1000 design, thus making the information's classification acceptable.

7.2.2.3.12 Operational and Maintenance Bypasses

Clauses 6.6, 6.7, 7.4, and 7.5 of IEEE Std. 603-1991 address the requirements as they relate to the operating and maintenance bypass of the sense and command and execute features of the safety I&C system. In part, the clauses require that operating bypasses automatically prevent the activation of an operating bypass or initiate the safety function if the applicable permissive conditions are not met. The capability of a safety system to accomplish its safety function shall be retained while equipment for sense and command and execute features is in maintenance bypass. Additionally, the guidance of RG 1.47 must be addressed to ensure the system responds appropriately in the presence of operational or maintenance bypasses.

The staff evaluated several sections of WCAP-16675-P regarding the PMS bypass capabilities. The four-division system has a high level of redundancy, which allows a division to be bypassed, placing the PMS logic function affecting a reactor trip or ESF actuation into a 2oo3 condition. Additionally, WCAP-16675-P, states that the design does not allow bypassing of two or more redundant channels or divisions. However, placing a given channel into a partial trip condition, if warranted, occurs automatically and does not require the "Function Enable" key switch on the MTP. The AP1000 DCD also commits to the prevention of the operating bypasses when permissive conditions are not met. This aspect of the design would be verified in the ITAAC described in AP1000 DCD Tier 1, Table 2.5.2-8, Design Commitments 9a–d and 11. Thus, based upon the information described above and that found in Section 7.5 of this report, the NRC finds the methodology described in WCAP-16675-P to bypass a given division of the PMS, for maintenance or operational conditions, to be acceptable.

7.2.2.3.13 Integrated Logic Processor Evaluation

The ILPs serve as the action-sequencing logic in the ESFAS portion of the PMS; they distribute the activate signal to the various CIMs, via their respective SRNCs. Based on the design information in WCAP-16675-P, the ILP acts as an intradivisional interface device between the comparative logic device (e.g., LCL) and the SRNC. The SRNC then forwards its output to the priority module (CIM). The ILP uses a design previously approved by the NRC (i.e., Common Q equipment). The staff identified this design information through examination of docketed material in Section 2.2.3.2.2 of WCAP-16775, which revealed the ILP uses an AC160 programmable logic controller to operate. As the AC160 is the processor within the Common Q platform, and the staff has previously approved its use, the staff finds the ILP design acceptable provided the remainder of its components function identically to that of a Common Q based module. The staff previously identified this issue as Open Item OI–SRP–7.2–ICE–04 and, based upon the review of the information in Section 2.2.3.2.2 of WCAP-16775, the staff finds the design of the ILP acceptable.

7.2.2.3.14 Component Interface Module Evaluation

The CIM serves as a transitional device that receives its safety-related input signals from the output of each of its respective SRNCs and delivers its output signal to the respective final actuation device of a given safety-related component. The CIM also serves as an interface device that receives input signals from the nonsafety-related PLS, in addition to the input signal it receives from the SRNC. The nonsafety-related communications (control) signal is applied to the CIM through the nonsafety-related RNC, which serves as a transitional device from the nonsafety-related to the safety-related systems.

The CIM system comprises the following major components:

Component Interface Module	(CIM)
Safety Remote Node Controller	(SRNC)
Double Wide Transition Panel	(DWTP)
Single Wide Transition Panel	(SWTP)
CIM Base Plates	
SRNC Base Plates	

The dual-redundant field programmable gate array (FPGA) based SRNCs receive their [ ] input from their respective ILP within the PMS via a high speed link (HSL). The SRNCs process the PMS signal [ ] prior to the safety-related PMS signal being presented to the CIM. Upon the completion of the signal conditioning being performed by the SRNC, the PMS signal is transmitted to the DWTP, which serves as a connection panel for various signals, and then to the CIM.

The DWTP connects the SRNC base plates and the incoming nonsafety-related Ovation® signal that is processed through the RNC to the input of the respective CIM. The DWTP also serves as the connection point for the 24 volts direct current (Vdc) power feeds that provide power to the SRNC and CIM base plates, the nonsafety-related RNC, and the SWTPs, if any are utilized. The SWTPs serves as extender modules that connect to the DWTP and allow additional CIMs to be connected to the given configuration of CIMs.

The SRNC and CIM base plates serve as physical mounting sockets to which a given SRNC or CIM can be attached.

The output signals from the DWTP or, if utilized, the SWTP, are presented to CIMs, namely the nonsafety-related Ovation® signal from the PLS and the dual redundant, safety-related PMS signal.

The CIM also utilizes FPGA programmable technology, which serves to evaluate and prioritize which of the two signals are present, and position the safety-related component accordingly.

IEEE Std. 603–1991 requires the CIM system satisfy all applicable criteria to ensure the device meets safety system requirements.

To address the single failure criterion in Clause 5.1 of IEEE Std. 603-1991, the applicant submitted WCAP-16438-P, Revision 3, and WCAP-16592-P, Revision 2. The AP1000 FMEA discusses the possibility of individual SRNCs and CIM failure modes and their effect upon the system. Sections 7.1.5, "I&C Architecture," and 7.2.2.3.2, "Single-Failure Protection," discuss the findings related to the PMS SHA and FMEA, respectively. The forthcoming FMEA for the CIM system will capture bounding failure types and the anticipated MTBF analysis for the components of the CIM system. Due to the forthcoming CIM system FMEA being bounded by the PMS FMEA (WCAP- 16438-P), the staff finds the approach of addressing the different failure modes of the CIM system adequate. Although both devices do not use software during their operational modes, the use of programming languages along with synthesizers, simulators and other testing tools being used during their lifecycles for both the SRNC and CIM require they be treated as software-based devices. Based upon the updated response to RAI-SRP-SHA-01, Revision 1, dated June 28, 2010, which discusses the CIM system within the SHA, the staff finds the CIM system satisfies the single failure criterion.

IEEE Std. 603–1991, Clause 5.2, requires the CIM system to be designed such that once the system has been actuated the executable features of the protective system continue to completion. Based upon the discussion in the AP1000 PMS FMEA (WCAP-16438-P), upon a failure of the SRNCs, such as through a loss of system power or overt module failure, the device will latch in its last command, such that if a failure occurs, the SRNC will continue to provide the actuate command. In the case of the CIM, should a single communication bus fail on the safety-related bus, the redundant bus will carry the signal through to completion. If a singular CIM fails outright, the system is designed with either a redundant CIM or another system design characteristic, such as a check valve in line with the valve actuator affected by the CIM failure to ensure the system is not adversely impacted by the component failure. These system design characteristics were verified during the April 20-22, 2009 audit conducted at the Westinghouse facility in Cranberry, Pennsylvania.

IEEE Std. 603–1991, Clause 5.3, along with 10 CFR Part 50, Appendix B, requires the CIM system to be designed with a sufficient measure of quality to ensure a high level of component and system reliability consistent with low failure rates. NUREG-0800 Appendix 7.1-D, "Guidance for Evaluation of the Application of IEEE Std. 7-4.3.2"; Section 5.3; and BTP 7-14 specifically address the criteria for a quality software development process. Additional regulatory guidance discussing acceptance criteria related to protective system quality are discussed in Section 7.2.2.3.3, "Quality." The staff found several areas of technical information to be lacking either a sufficient level of detail to satisfy BTP 7-14 or a proposed alternative method of providing adequate CIM system developmental detail in order for the staff to find the CIM development process of sufficient breadth and technical depth. As such, the final disposition of this issue related to CIM development, previously addressed as Open Item OI-SRP7.2-ICE-05, will be covered in Section 7.2.5.

IEEE Std. 603-1991 Clause 5.4, "Equipment Qualification," states, in part, that safety system equipment shall be qualified by type test, previous operating experience, or analysis, or any combination of these three methods, to substantiate that it will be capable of meeting, on a continuing basis, the performance requirements as specified in the design basis. Based upon a review of the ITAAC in Table 2.5.2-8, particularly Design Commitments 2, 3, 4, 5a and 5b, the listed commitments address the protective systems; therefore, the CIM system's ability to withstand the necessary seismic, electromagnetic/radio-frequency, and electrostatic discharge loads or transients. Additionally, the CIMs must be able to withstand additional environmental conditions, such as qualified temperature and humidity profiles. As a result of the requirements placed upon the protective system and, therefore, the CIM system, in the Tier 1 ITAAC of Section 2.5.2 of the AP1000 DCD, the staff finds that the CIM system meets the environmental qualifications as required by IEEE Std. 603-1991.

Safety systems must also be capable of accomplishing their safety functions under the full range of applicable conditions specified in the design basis in accordance with Clause 5.5, "System Integrity," of IEEE Std. 603-1991. Additionally, the staff used the guidance in Appendices 7.1-C and 7.1-D of NUREG-0800 to review the CIM system and the manner by which it meets the requirements of system integrity.

Of particular concern when dealing with computer-based or programmable technology-based systems is the system's ability to complete its function in an actual or real time period that would ensure that any design basis limits are not exceeded. The system must be left in a known state after an event condition has been resolved to satisfy Information Bulletin 80-06; "Engineered Safety Features (ESF) Reset Controls," thus ensuring that it is the operator, not the system that

controls the repositioning of equipment during recovery from an event. Additionally, a digital system should be able to detect, alert the operator, and take appropriate action, perhaps by taking a component, channel or division to trip or bypass, once a failure is detected.

Based upon a review of WCAP-16674-P, WCAP-16675-P, and Section 2.3.1.2.8 of WCAP-17179-P, the staff determined the CIM subsystem portion of the PMS system functioned in a deterministically-based manner. This information is also captured in lower level proprietary Westinghouse documents, thus helping to ensure safety-system timing requirements are validated. Based upon Section 6.0, "Component Interface Module," of WCAP-16674-P, the applicant committed that, after the completion of an event scenario, the PMS components will remain in their actuated state until they can be repositioned by an operator. This commitment adequately addresses the guidance of Information Bulletin 80-06. In relation to the guidance of Appendix 7.1-C of NUREG-0800 detailing a protection system's response in the presence of a detected failure of a component, channel or division, the staff finds the information in Section 2.1 of WCAP-17179-P adequately addresses the system's self-identification and reporting of system faults. In addition, the FMEA for the PMS (WCAP-16438-P) discusses the overall system response in the presence of known component, channel or system faults in Section 4.4 ILC, "(Integrated Logic Cabinet) Process Station." Based upon a review of the docketed information presented by the applicant, related to system integrity, the staff concludes the CIM system operates in a manner consistent with Clause 5.5 of IEEE Std. 603-1991.

IEEE Std. 603-1991, Clause 5.6, "Independence," along with 10 CFR Part 50, Appendix A, GDCs 21, 22, and 24, require, in part, that safety-related protection systems be designed with sufficient physical, electrical, functional, and communications independence. The communications independence of the CIM will be discussed in Section 7.9. Electrical isolation of the CIM system is provided through the use of fiber optic connections from the PLS, through the RNC, to the DWTP within the ILC, in which the CIM system resides. Physical independence is provided by housing all CIM system safety-related components separately from the nonsafety-related PLS components, with one notable exception. The RNC is housed in the ILC with the CIM system. Although not classified as a Class 1E component, the applicant committed to qualify the RNC as an associated circuit in accordance with IEEE Std. 384-1981. Although the functionality of the RNC cannot be guaranteed during or after a design basis event, the applicant's commitment to qualify the RNC as an associated circuit prevents the loss of the CIM's safety function in the event of an RNC hardware failure. As such, the interaction of the PLS and PMS within the CIM also addresses functional isolation in that a loss of function within the RNC or another PLS-based component will not inhibit the CIM from carrying out its safety-related function in accordance with 10 CFR Part 50, Appendix A, GDC 24. For further information related to PMS independence, refer to Section 7.2.2.3.6, "Independence."

Based upon a review of the information in WCAP-16674-P, Revision 4; WCAP-16675-P, Revision 5; and WCAP-17179-P, Revision 2, the staff finds that the CIM system meets the criteria related to physical, electrical and functional independence requirements between the safety-related and nonsafety-related systems. A discussion related to the findings on communications independence of the CIM system is provided in Section 7.9.

IEEE Std. 603-1991, Clause 5.7, and 10 CFR Part 50, GDC 21, require the protection system be capable of testing and calibration during all anticipated modes of operation. In addition, the guidance in Appendix 7.1-D and BTP 7-17, of NUREG-0800, as well as RG 1.22, state how a system designed for high reliability will be able to be tested in various configurations while simultaneously ensuring the protection system stays in service. Based upon a review of all applicable information in Chapter 7 of the AP1000 DCD and associated TRs, the applicant

credits no self tests that mitigate or replace the need to conduct periodic surveillance testing of the CIM or protection systems. After reviewing the CIM system TR, the system conducts periodic diagnostic self-tests and transmits the information via the CIM and SRNC to the ILP within the PMS, at which point the Common Q portion of the system alerts the operator to any condition outside of normal system operation. This topic is discussed further in Section 2.5 of the CIM TR. The staff finds that based upon a review of the information in WCAP-16674-P, Revision 4; WCAP-16675-P, Revision 5; and WCAP-17179-P, Revision 2; the CIM system satisfies Clause 5.7 of IEEE Std. 603-1991 and 10 CFR Part 50, Appendix A, GDC 21.

Clause 5.9, "Control of Access," in IEEE Std. 603-1991, stipulates appropriate levels of control be required to access the safety system. The PMS is typically accessed from either the PDSP in the MCR or via the division specific MTP. The CIM system components can be accessed when required, such as is the case when safety-related components with non-onerous consequences cannot be controlled via the PLS via the local controls on the CIM itself, but the cabinets in which the CIMs are housed are locked and administrative controls will be in place to prevent unwanted access of the CIM system. A more detailed analysis concerning the control of access to the CIM and PMS is provided in Section 7.9.5, which evaluates the applicant's requirements to ensure that the PMS is constructed in an SDOE.

Based upon a review of the material in WCAP-17179-P, Revision 2, in particular Section 2.12, "Human Factors and Maintenance Considerations," the staff finds that the CIM system adequately addresses Clause 5.10, "Repair," of IEEE Std. 603-1991.

In relation to the use of automatic and manual control, the CIM system is not designed for routine manual control. However, in the event of the failure of a safety-related component with non-onerous consequences that does not have individual safety-related controls on the safety-related PDSP in the MCR (as defined by the PMS FMEA in WCAP-16438-P, Revision 3), the CIM possesses local safety-related controls that can be utilized to position the safety-related component as the situation requires. It should be noted these non-onerous components are typically soft-controlled from the PLS; however, as the PLS is nonsafety-related, its actions are not credited in accordance with IEEE Std. 603-1991. Further discussion regarding how the PMS executes automatic and manual controls may be found in Section 7.2.2.3.11 of this report. Based upon its review of WCAP-17179-P, Revision 2, the staff finds that the CIM system meets the requirements of IEEE Std. 603-1991, Clauses 6.1 and 6.2, "Automatic," and "Manual Control," respectively.

IEEE Std. 603-1991, Clause 5.15, requires the protection system operate in a reliable manner. To facilitate this requirement as it relates to the CIM system, appropriate analyses shall be conducted to ensure that reliability goals have been achieved. The applicant has committed to present to the staff a formal CIM development process, in which it will demonstrate what quality measures, processes, policies, procedures and analyses will be completed in order to satisfy the requirements of IEEE Std. 603-1991, Clause 5.15. Additional discussion of the staff's understanding of the applicant's commitments made related to CIM system reliability as a byproduct of the CIM development process, may be found in Section 7.2.5 of this report.

The staff conducted an engineering review with the applicant's technical personnel on October 15-16, 2008, and January 29-30, 2009, and conducted two audits, one on April 20-22, 2009, and the other on April 12-16, 2010, to discuss the PMS development process, of which the CIM is a critical part. Previously through the use of Open Item OI-SRP7.2-ICE-05, the staff raised the issue of an inadequate CIM development process and the staff restated this conclusion to the applicant after the April 12-16, 2010 audit and transmitted RAI-SRP7.0-ICE-11

to the applicant on May 17, 2010. After reviewing the updated response to the RAI in WCAP-17179, the staff found that the applicant did not provide sufficient planning or design information related to the CIM priority module. The following provides a topical breakdown of the required information:

- Section 5.1.5 of WCAP-16675-P describes the CIM as a non-software-based Class 1E device that is not considered to be susceptible to a software common-cause failure (CCF). However, the CIM is FPGA-based (programmable technology). The applicant did not provide sufficient information for the staff to determine that the CIM is not susceptible to a software common-cause failure (SWCCF). The applicant committed to updating all TRs by removing all text stating the CIM is not susceptible to a SWCCF.

- The applicant has not adequately described the programmable technology lifecycle development plans and processes for the CIM logic development, design, implementation, testing, and operation.

The applicant has not provided sufficient information on these topics, previously requested under RAI-SRP7.1-ICE-21, RAI-SRP7.1-ICE-22, and RAI-SRP7.1-ICE-23. Additionally, after conducting two audits, one in Scottsdale, Arizona (at the CIM and DAS supplier's facility) and the other in Warrendale, Pennsylvania (Westinghouse's location) and reviewing the information in WCAP-17179, the required information, in both depth and breadth, was determined not to be present for the CIM or its peer components in the AP1000 CIM TR. The staff previously identified this issue as Open Item OI-SRP7.2-ICE-05, and, based upon the applicant's commitment to develop an adequate CIM development process; this open item will be closed out as discussed in Section 7.2.5.

The additional detailed design information would otherwise have to be addressed through verification of implementation of the I&C DAC. Therefore, the changes to the DCD eliminate the need for I&C DAC and, thus, satisfy the finality criteria in 10 CFR 52.63(a)(1)(iv).

7.2.2.3.15 Safety-Related Remote Node Controller Evaluation

The staff conducted an audit that dealt with the review of the Phase 1 and Phase 2 proprietary documents for the AP1000 PMS SLC on April 20–22, 2009, in Cranberry, Pennsylvania. During the demonstration of a "test" system, the staff learned of a new device that the applicant would add to the PMS. The applicant demonstrated the use of an SRNC that would serve as the interface device from the ILP to the CIM. Under previous revisions of TRs, no intermediary device existed between the ILP and the CIM. Additionally, the staff reviewed WCAP-17179-P, the AP1000 CIM TR that discussed the use of the SRNC within the CIM system. Based upon the review of the report, questions regarding the use of and quality design process built into the SRNC device remain. Therefore, the issue was captured under the open item related to the CIM. The issue previously identified as Open Item OI-SRP7.2-ICE-06 pertaining to the quality development process of the SRNC will be captured by the response to Open Item OI-SRP7.2-ICE-05, which discusses quality development process of the CIM, of which the SRNC is a part. This open item is discussed in Section 7.2.5.

7.2.3 Common Qualified Platform Design and COL Action Items

In TR-42, the applicant stated its position on what current and future activities it will complete through the ITAAC process. In some cases, the closeout activities either point to more than one ITAAC for closure, or the applicant requested closure of a given PSAI when it had not met all the acceptance criteria for a PSAI. The information in TR-42 is not entirely consistent with the staff's position regarding the disposition of the GOIs and PSAIs.

To identify the applicant's position regarding currently open and previously closed PSAIs and GOIs through the ITAAC process, the NRC asked the applicant, in RAI-SRP7.1-ICE-01, to provide a detailed road map showing which I&C design items remain open and which have already been closed, as well as the method of closure through the ITAAC or DAC process.

On December 12, 2008, the NRC received the applicant's response to RAI-SRP7.1-ICE-01 in DCP/NRC2319. However, several of the applicant's conclusions were not consistent with the staff's findings, which are based on a review of the information in the following documents:

- "Safety Evaluation by the Office of Nuclear Regulation CE Nuclear Power Topical Report CENPD-396-P Common Qualified Platform Project 692," and its supporting future documents regarding the closeout of GOIs and PSAIs, also known as Category 1 and Category 2 Closeout Items

- APP-GW-GLR-017

- Response letter to RAI-SRP7.1-ICE-01, dated December 12, 2008

The staff determined that GOIs 7.1, 7.2, 7.3, 7.5, and 7.6 are closed, based on the findings in "Acceptance of the Changes to Topical Report CENPD-396-P, Revision 1, 'Common Qualified Platform,' and Closeout of Category 2 Open Items (TAC No. MB2553)." The NRC closed out GOIs 7.4 and 7.7 generically, as well as PSAI 6.3. The NRC closed GOI 7.9 and 7.10 in acceptance of "design concept only," based on findings in "Safety Evaluation for the Closeout of Several of the Common Qualified Platform Category 1 Open Items Related to Reports CENPD-396-P, Revision 1, and CE-CES-195, Revision 1 (TAC No. MB0780)."

The following provides the staff's position with regard to the remainder of the open GOIs and PSAIs:

- GOI 7.8

  The staff reviewed WCAP-16674, Revision 4. The staff determined the methodology used by the applicant that the staff understands to be a "safe state, system-based" approach, which typically used the nonsafety-related PLS to control a safety-related component, was acceptable, in that the system first attempts to activate the given safety-related component using the PLS and, failing that, the system then activates using the safety-related PMS. In the event of a SWCCF of the PMS, the AP1000 compensates by activating the DAS to meet the diversity requirements in BTP 7-19. Although other issues related to the development process of the priority device, known as the CIM system in the AP1000 design, still remain, the mechanism by which the safety-related component actuates, either by a nonsafety-related or safety-related system means, is deemed acceptable. Therefore, the staff considers GOI 7.8 closed.

However the issue of overall CIM system quality, planning and development processes continues to be a concern previously captured under Open Item OI-SRP7.2-ICE-05 and, based upon the applicant's commitment to adequately address a high quality CIM development process. This action is discussed in Section 7.2.5.

- GOI 7.9

  The applicant stated in TR-42 that the NRC can close GOI 7.9, regarding the specific use of the ITP and the Advant Fieldbus (AF) 100 buses to provide separation of safety and nonsafety signals, since the AP1000 I&C system differs in some details from the integrated solution described in Appendix 4, "Common Qualified Platform Integrated Solutions." Additionally, this TR provides other plant-specific implementations of safety-to-nonsafety communications.

  Thus, the staff considers GOI 7.9 closed. The staff evaluated the other plant-specific implementations of safety-to-nonsafety communications in Section 7.9.

- GOI 7.10

  The applicant committed to alter the characteristics of the AP1000 Common Q platform so that the AP1000 design no longer uses multichannel operator stations. Therefore, the NRC considers GOI 7.10 closed.

- PSAI 6.1

  The staff agrees with the applicant's conclusion in TR-42 that it uses its quality assurance program to determine the suitability of all I/O devices, by following the requirements of the Westinghouse QMS, which the NRC approved in August 2002. The NRC considers PSAI 6.1 closed.

- PSAI 6.2

  The staff agrees with the applicant's conclusion in TR-42 that, since the AP1000 does not use a hardware user interface, PSAI 6.2 is not valid in the AP1000 specific application. Therefore, the NRC considers PSAI 6.2 closed.

- PSAI 6.3

  The NRC closed out this issue (see above discussion).

- PSAI 6.4

  The staff agrees with the applicant's conclusion in the RAI response letter dated December 12, 2008 (see above), as well as the information in "Resolution of Common Q NRC Items for AP 1000," Revision 0, also referred to as TR-42, that this PSAI will not be closed until the completion of the testing phase for hardware and software for the PMS. This commitment to test the system's components to appropriate levels of environmental qualification appears in AP1000 DCD Tier 1, Chapter 2, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 4.

- PSAI 6.5

  The staff agrees with the applicant's conclusion in the RAI response letter dated December 12, 2008 (see above), that this PSAI will be addressed in the testing phase for PMS hardware and software. The commitment to verify the implementation of the SLC appears in AP1000 DCD Tier 1, Chapter 2, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 11.

- PSAI 6.6

  The staff agrees with the approach the applicant selected regarding the AP1000 setpoint methodology in WCAP-16361-P, Revision 1. AP1000 DCD Tier 1, Chapter 2, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 10, addresses the actual setpoint accuracy and response time of AP1000 safety systems. Since the DCD provides an acceptable setpoint methodology and ITAAC to verify setpoints and response time, the NRC considers PSAI 6.6 closed. Section 7.2.7 includes further discussion of setpoints.

- PSAI 6.7

  The staff agrees with the applicant's conclusion in the RAI response letter dated December 12, 2008 (see above), that this PSAI will be addressed during the human factors engineering (HFE) testing phase for the PMS. AP1000 DCD Tier 1, Chapter 3, ITAAC Table 3.2-1, includes the design commitments for verifying the HFE. Further information is provided in Chapter 18 of NUREG-1793.

- PSAI 6.8

  The staff agrees with the applicant's conclusion in the RAI response letter dated December 12, 2008 (see above), that this PSAI is applicable to existing NPPs, not new power plants incorporating new designs. Therefore, the NRC considers this PSAI closed.

- PSAI 6.9

  The staff agrees with the applicant that the plant procedures and/or technical specifications due to installation of the Common Q system will be dealt with at the plant-specific level. Chapter 16 of NUREG-1793 and this supplement address the technical specifications.

- PSAI 6.10

  Previously, the staff reviewed but had not approved the generic FMEA, and submitted several RAIs to the applicant to offer a more detailed technical response or to clarify several statements in the FMEA document that were unclear. Section 7.2.2.3, "Single-Failure Protection," which discusses the technical information required of the applicant related to its FMEA. As a result, the NRC considered this PSAI open. Based upon the submission of Revision 3 of WCAP-16438-P, the PSAI is considered closed. For further information, regarding this topic, refer to Section 7.2.2.3.2, Single Failure Protection.

- PSAI 6.11

  PSAI 6.11 states that an applicant using the Common Q platform would need to address D3 to prevent an SWCCF. In Sections 7.1 and 7.7 of the AP1000 DCD, the applicant describes the functional requirements of the DAS. The DAC in Revision 19 of AP1000 DCD Tier 1, Table 2.5.1-4, Item 4, includes the design and analysis of the DAS. As described in Section 7.8, the applicant modified Item 4. The staff found that the applicant provided sufficient design information to justify the modifications. Therefore, this PSAI is closed.

- PSAI 6.12

  AP1000 DCD Tier 1, Chapter 2, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 10, defines the commitment to verify proper response times of circuits. Since the applicant has provided an acceptable ITAAC to address this action item, the NRC considers PSAI 6.12 closed. The time response of the CIM system is addressed in Section 7.2.2.3.14 of this report.

- PSAI 6.13

  The staff agrees that this PSAI will remain open until the completion of integration and preoperational testing. AP1000 DCD Tier 1, Chapter 2, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 11, includes the commitment to verify load capacity and sharing of communications resources for the PMS.

- PSAI 6.14

  PSAI 6.14 states implementation of the Common Q must not render invalid any previously accomplished Three-Mile Island (TMI) action items. In NUREG-1793, the staff found the AP1000 design addresses the TMI action items. The staff did not find any information in the proposed changes to the AP1000 DCD that would invalidate that conclusion. Since the AP1000 design meets the I&C-related TMI action items described in 10 CFR 50.34(f)(2), "Contents of applications; technical information," the staff finds this PSAI closed.

Based upon the discussions above, the staff considers all GOIs closed. The open PSAIs listed above will continue to remain open until they are resolved.

7.2.5 Protection and Safety Monitoring System Design Process Review

In AP1000 DCD Tier 1, Section 2.5.2, the applicant describes its entire SLC process related to the development of the Common Q portion of the PMS, which it will implement during the planning, design, construction, testing, and operational phases for the AP1000 I&C safety systems. In AP1000 DCD Tier 1, Section 2.5.2, Design Commitment 11, the applicant deleted the design requirements phase and system definition phase. The applicant based this removal on the cumulative amount of both docketed and audited documentation made available to the staff as of this date.

The staff reviewed the information on the docket and conducted several site visits related to the review of the PMS design process at Westinghouse's Twinbrook facility, and in both Monroeville and Cranberry, Pennsylvania. The primary purpose of the Twinbrook visits (April 8–10, 2008, October 15–16, 2008, January 29–30, 2009, and July 30, 2009) was to conduct an engineering review of the documents for the design requirements and system definitions phases (described as the conceptual phase and system definition phase by the Common Q SPM and WCAP-15927, and the planning activities and requirements activities phases of NUREG-0800 BTP 7-14 SLC Process).

Additionally, the staff conducted three audits, the first was conducted on April 20-22, 2009, examining the Phase 1 and Phase 2 AP1000 PMS SLC proprietary material at the Westinghouse facility in Cranberry, Pennsylvania. The second audit was conducted on March 8–11, 2010, at CS Innovations, in Scottsdale, Arizona. CSI serves as the designer and supplier of the CIM subsystem (as well as the DAS – Section 7.8 provides further discussion of how adequate diversity is maintained between the two systems) within the PMS. The more recent audit was conducted in Warrendale, Pennsylvania at the Westinghouse Automation Services (formerly Repair, Replacement, and Automation Services) facilities on April 12-16, 2010. The more recent of the two audits was in order to determine whether an adequate demonstration of documentation had been presented to conclude that all requirements for the planning activities phase and requirements activities phases for both the PMS and the DAS would be considered complete. The October 3-5, 2006, trip report lists the design requirements phase documents associated with the staff's visit to the Monroeville, Pennsylvania, facility as of October 2006. The applicant based its conclusion that its design requirements and systems definition phases were complete on the proprietary information listed in the April 2009 audit report and the docketed information related to the AP1000 I&C safety systems design process. The applicant desired to close these two phases as part of its DAC closure process.

The staff finds that, once the requirements for each phase of the PMS (SLC) are met, "completion" rather than "elimination" of these and all phases described in the text of AP1000 DCD, Tier 1, Section 2.5.2, Item 11, is appropriate, provided the staff finds the information in those phases to be sufficient. When the staff arrives at that conclusion for each given design process phase, The applicant may remove the given SLC phase(s) in Tier 1, Table 2.5.2-8, Item 11. Those tables describe specific ITAAC activities that will be completed during the given facility's inspection process, rather than the process undertaken to ensure that the applicant has included sufficient quality in the overall design process for AP1000 I&C safety systems. However, the applicant may not remove the design process description from the text-based portion of the design process description in Tier 1, Section 2.5.2, Item 11. The applicant agreed to restore all AP1000 PMS design process phases to the text-based portion of Tier 1, Section 2.5.2, Design Commitment 11. On February 23, 2010, the applicant submitted a response to Open Item OI-SRP7.2-ICE-07 that dealt with the removal of the text based

descriptions in Tier 1, Section 2.5.2. The applicant's commitment to reinsert the text-based portions of the PMS design process with the addition of the word "complete" to those given phases, once the staff finds a given developmental phase for the PMS to be adequately addressed, is an approach acceptable to the staff. However, although not specifically addressed by this open item, the staff's expectation would be that this process would be carried out in the Tier 1, Section 2.5.1 for the DAS in a similar fashion. The staff previously identified this as Open Item OI-SRP7.2-ICE-07. In a subsequent revision, the applicant incorporated this Tier 1 information in the AP1000 DCD. The staff concludes that these changes are acceptable.

Based on the review of all audited and docketed documentation provided to the NRC that relates to the design requirements, system definition, and remaining developmental phases of the SLC for the PMS, as well as the series of audits conducted and their conclusions, the staff has not concluded that the applicant adequately completed or addressed the system definition phase, both from a lack of technical adequacy and a lack of completeness of the given subject matter. After reviewing Section 7.1.2.14.1, "Design Process," of the AP1000 DCD, the staff was unable to locate additional information adequately describing all developmental phases of the Common Q SLC and the programmable technology lifecycle for the CIM system in Tier 2 of the AP1000 DCD. Additionally, based upon discussions held during the April 2009, March 2010, and April 2010, audits and conclusions drawn in the respective audit reports, the applicant stated they had "split" the system definition phase of development for the Common Q and CIM system portion of the PMS. However, based upon the documents made available for review, it appears the applicant has "split" the system definition phase of development in a discussion-based format only, as no alteration to the Common Q SPM, the Tier 2* document, WCAP-15927, or Tier 2 information has been made. The applicant did not provide sufficient technical information in the AP1000 DCD, its associated TRs, or its proprietary documentation (to be made available for audit or inspection) to demonstrate satisfactory completion or alteration of the system definition phase of Common Q development.

The applicant must provide adequate information regarding a programmable technology development plan for the CIM and SRNC, previously addressed by Open Items OI-SRP7.2-ICE-05 and OI-SRP7.2-ICE-06, respectively. The applicant must restore the listing of the system definition phase in AP1000 DCD Tier 1, Section 2.5.2, Design Description 11, and in ITAAC Table 2.5.2-8, Design Commitment 11, until the staff finds that the applicant has completed that group of activities. Once those activities had been completed, the staff required the applicant to comply with its commitment as delineated in its response to Open Item OI-SRP7.2-ICE-07, which dealt with the removal and restoration of the text-based descriptions in Tier 1, Section 2.5.2. Additionally, the staff required the applicant to restore information in Tier 1, provide additional, sufficient information in Tier 2, and Tier 2* documentation, especially WCAP-15927, that accurately describes all developmental phases and processes associated with the Common Q portion of the PMS. The staff previously identified this issue as Open Item OI-SRP7.2-ICE-08. Based upon the applicant's commitment to restore the system definition phase of the PMS SLC in a future revision of the AP1000 DCD and based upon the applicant's commitment to add ITAAC Design Description 14, in Tier 1, Section 2.5.2, and Design Commitment 14 in Table 2.5.2-8, in which the applicant describes how it will meet the requirements related to the development of the CIM subsystem within the PMS, the staff determined that Open Items OI-SRP7.2-ICE-05 and OI-SRP7.2-ICE-08 were closed. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Since the applicant chose to add a more specific ITAAC related to the CIM development process (Tier 1, Section 2.5.2, Design Description 14 and Design Commitment 14 in

Table 2.5.2-8, ITAAC), and based upon the additional language added to the system integration and test phase acceptance criteria in Design Commitment 11, as discussed in Section 7.9.2.3.2 of this report, the staff finds the design requirements, or planning activities phase (in accordance with HICB BTP-14), of development for the PMS to be complete. However, the staff's expectation related to the forthcoming CIM development process is that it will sufficiently address and describe all phases of development for the CIM subsystem of the PMS, including the planning activities phase of development.

Furthermore, since the applicant submitted, and the staff approved, critical licensing documents in the system definition phase related to PMS development (such as the FMEA for the AP1000 PMS (WCAP-16438-P) and the SHA of the AP1000 PMS (WCAP-16592-P)), the staff does not consider the remaining development activities listed as part of the system definition phase of the PMS SLC to be DAC, as future development activities are not anticipated to impact licensing basis information, such as the AP1000 DCD Tier 2 or AP1000 TRs referenced in the DCD. However, if the design detail from future development activities impacts the licensing basis information, the staff expects that information to be incorporated into the licensing basis information.

In Revision 16 of the AP1000 DCD, the applicant asked to remove the reference to WCAP-15927, which the applicant submitted to the NRC in addition to the SPM, to resolve RAI 420.001 and RAI 420.023. The staff issued these RAIs during the certification of the original NUREG-1793 in 2002. The applicant had to demonstrate the measures it would take to ensure critical information in this report is not removed. In July 2009, the applicant decided not to remove the report and submitted Revision 2 of WCAP-15927, which explained which organization, in this case, the separate and independent IV&V organization, has the exclusive responsibility for IV&V activities, including testing activities. The staff considers the issue resolved, since WCAP-15927 elaborates on the organization that performs the software verification and validation (V&V) activities. Specifically, Section 3 of the document states that "...testing activities are defined as part of the V&V process." The statement indicates that the IV&V group is responsible for testing the PMS, as discussed in the Common Q SPM. Therefore, the staff finds that RAI-SRP7.1-ICE-10 is resolved.

The additional detailed design information would otherwise have to be addressed through verification of implementation of the I&C DAC. Therefore, the changes to the DCD eliminate the need for I&C DAC and satisfy the finality criteria in 10 CFR 52.63(a)(1)(iv).

7.2.7 Protection Systems Setpoint Methodology

On May 30, 2006, the applicant submitted WCAP-16361-P, Revision 0. The following regulatory requirements and guidance documents apply to the staff's review of WCAP-16361-P:

- GDC 13 and 20
- 10 CFR 50.36(c)(ii)(A), "Technical specifications," requires that the technical specifications include limiting safety-system settings
- RG 1.105, "Setpoint for Safety-Related Instrumentation," describes a method acceptable to the staff for complying with the NRC's regulations for ensuring that setpoints for safety-related instrumentation are initially within, and remain within, the safety limit

The Westinghouse setpoint methodology combines the AP1000 uncertainty components to determine the overall channel statistical allowance for the functions of the RTS/ESFAS. All appropriate and applicable uncertainties, as defined by a review of the AP1000 baseline design input documentation, have been considered for each RTS/ESFAS function. The methodology used to combine the uncertainty components for a channel is an appropriate combination of those groups that are statistically and functionally independent. Those uncertainties that are not independent are conservatively treated by arithmetic summation and then systematically combined with the independent terms. It includes instrument (sensor and process rack) uncertainties and non-instrument-related effects (process measurement accuracy). The methodology used the square-root-of-the-sum-of-the-squares technique, which the NRC has approved. Also, the American National Standards Institute (ANSI), the American Nuclear Society (ANS), and the International Society of Automation (ISA) approve the use of the same probabilistic and statistical techniques for the various standards that determine safety-related setpoints.

The staff reviewed WCAP-16361-P and found that the allowable values (AVs) are equal to the rack calibration accuracy, which is the acceptable "as-left" value. This methodology ensures that the purpose of the AV is satisfied by providing a large enough allowance to account for those uncertainties not measured during the surveillance tests to protect the safety limit. Also, the difference between the AV and the nominal trip setpoint is as large as the calibration tolerance, and the AVs, along with the nominal trip setpoint, are included in the plant technical specifications as the associated criteria, in accordance with 10 CFR 50.36. Therefore, the staff concludes that the proposed WCAP-16361-P, Revision 0, as it relates to an overall setpoint methodology, is acceptable. However, due to proposed changes in the inputs to the OPΔT and OTΔT as discussed in Section 7.2.2.1.1 of this report, the applicant committed to revise the WCAP-16361-P report to address these changes. These changes were incorporated into Revision 1 to WCAP-16361-P, which the staff reviewed and found acceptable.

Section 7.1.6.1 of the AP1000 DCD states that all requested information on the subject of setpoint methodology and final setpoint calculations has been completely addressed and requires no further action by the COL applicant. This statement is not in agreement with WCAP-16361-P, in which the applicant concludes that it cannot determine the final setpoint calculations until it completes the final design of the power plant. The staff issued RAI-SRP7.2-ICE-08, requesting that the applicant demonstrate how it intends to meet the final calculation requirements, given that it has not completed the protection system design. The applicant submitted DCR/NRC2315 to the NRC on December 9, 2008, declaring that the COL applicant will determine the setpoint adequacy, in accordance with AP1000 DCD Tier 1, Table 2.5.2-8, Item #10. In its response to Open Item OI-SRP7.2-ICE-09, dated March 8, 2010, the applicant committed to restore the language in Section 7.1.6.1, stating that the COL applicant or licensee will be responsible for the final determination of setpoints, in accordance with AP1000 DCD Tier 1, Section 2.5.2, ITAAC Table 2.5.2-8, Design Commitment 10. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Section 2.2.6 of WCAP-16675-P states that the applicant uses the MTP to alter setpoints and addressable constants. The MTP provides a dedicated display interface for each division and is used to bypass the channel before changes are made. The staff evaluated the access controls on the MTP to control the alteration of addressable constants, setpoints, parameters, and other settings to meet the requirements of IEEE Std. 603-1991, Clause 5.9. This clause required the design to provide administrative control of access to safety-system equipment. Section 1, Point 10, of the digital I&C ISG Document #4, "Highly-Integrated Control Rooms—

Communications Issues (ISG #4-HICRc)," Revision 1, clarifies this requirement by making the following statement:

> A workstation (e.g., engineer or programmer station) may alter addressable constants, setpoints, parameters, and other settings associated with a safety function only by way of the dual-processor/shared-memory scheme described in this guidance, or when the associated channel is inoperable. Such a workstation should be physically restricted from making changes in more than one division at a time. The restriction should be by means of physical cable disconnect, or by means of keylock switch that either physically opens the data transmission circuit or interrupts the connection by means of hardwired logic.

The staff finds the use of the MTP acceptable to bypass the channel in addressing the guidance in ISG No. 4-HICRc. Specifically, since the MTP serves as a dedicated display interface system for each division, the NRC does not require physical means to prevent the MTP from making changes to more than one division at a time.

7.2.8 Protection and Safety Monitoring System Evaluation Findings and Conclusions

The staff evaluated the revisions made to Chapter 7 of the AP1000 DCD, against the regulatory requirements stipulated in NUREG-0800 Table 7-1. Below is a summary of the staff's findings as they relate to the AP1000 PMS.

The staff finds that the proposed changes to the AP1000 DCD meets the requirements of 10 CFR 50.55a(a)(1). Specifically, the staff found that the applicant incorporated quality standards into the design of the AP1000 I&C systems.

Regulations in 10 CFR 50.55a(h) require compliance with IEEE Std. 603-1991 and the correction sheet dated January 30, 1995. The staff compared the PMS in the amendments to the AP1000 I&C systems design with the applicable clauses of IEEE Std. 603-1991 and has the following findings:

- The applicant satisfied Clause 5.1 of IEEE Std. 603-1991, as documented in Section 7.2.2.3.1.1. Specifically, the applicant has demonstrated how the PMS met the criteria presented in Clause 5.1 and GDC 21. The FMEA provided to the staff adequately demonstrates how the PMS will operate with a single failure under all postulated operating conditions, as discussed in Section 7.2.2.3.2 of this report.

- IEEE Std. 603-1991, Clause 5.3: The applicant satisfies the requirement of quality for the PMS, as documented in Section 7.2.2.3.3. Specifically, the applicant has not demonstrated how it meets the criteria in Clause 5.3 with regard to the design of the CIM and the SRNC, which would then provide reasonable assurance that it had developed a high-quality product for all components within the PMS. Sections 7.2.2.3.14 and 7.2.2.3.15 of this report discuss these issues. The staff previously identified these issues as Open Items OI-SRP7.2-ICE-05 and OI-SRP7.2-ICE-06, respectively. Based upon the applicant's commitment to add an ITAAC relating to the development of a programmable technology lifecycle for the CIM system, as demonstrated by the addition of Design Description 14 and Design Commitment 14 to Table 2.5.2-8 of the DCD, that is consistent with the requirements of IEEE Std. 603-1991 and the guidance of BTP 7-14 of NUREG-0800, the staff considers the open items closed.

10 CFR Part 52, "Licenses, certifications, and approvals for nuclear power plants," Appendix D, "Design Certification Rule for the AP1000 Design," Section II.D, defines Tier 1 information as that information explaining design descriptions, along with ITAAC information. Based upon the applicant's commitment to restore all phases of the PMS SLC design process in the text-based portion of Item 11 in NUREG-0800 Section 2.5.2, "Protection and Safety Monitoring System," the issue the staff previously identified as Open Item OI-SRP7.2-ICE-07 is closed.

- 10 CFR 52.47(b)(1) describes the ITAAC. The applicant provided sufficient information to satisfy the completion of the design requirements or conceptual phase of the PMS SLC, with the commitment to add the CIM Development Process ITAAC discussed above. Based upon its review of the documentation presented and based upon conclusions drawn in several audits, the staff will not approve the removal of the system definition phase of the PMS SLC until such time as the applicant provides satisfactory information to the staff for review and approval.

- IEEE Std. 603-1991, Clause 6.8: The staff found the proposed setpoint methodology acceptable. Additionally, the applicant commitment to restore the text in Section 7.1.6.1 of the AP1000 DCD stating that the COL applicant or licensee will be responsible for the final determination of setpoints was found acceptable to the staff. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff finds that due to the addition of the ITACC related to the information that will be provided for the CIM development process, (design description and Design Commitment 14 in Tier 1, Chapter 2.5.2 and Table 2.5.2-8, respectively) the design requirements or planning activities phase of development for the PMS is considered complete.

The staff finds that the proposed changes to the AP1000 DCD did not affect the remaining requirements of IEEE Std. 603-1991. Therefore, the staff's original conclusions in NUREG-1793 for these requirements are still valid.

The staff finds that proposed changes to the AP1000 DCD did not present changes that would invalidate the staff's conclusions in NUREG-1793 regarding the requirements in 10 CFR 50.34(f)(2).

Appendix A to 10 CFR Part 50 provides specific criteria for the I&C systems. The staff found that the PMS design continues to comply with the specific GDC for the I&C systems in NUREG-0800 Table 7-1, as described in NUREG-1793.

Regulations in 10 CFR 52.47(a)(9), "Contents of applications; technical information," require that applications for light-water-cooled NPPs evaluate the standard plant design against NUREG-0800 revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in NUREG-0800 acceptance criteria. NUREG-0800 Chapter 7 provides the design considerations for safety, including criteria for performance and reliability considerations. The staff evaluated the information presented for AP1000 safety systems in the AP1000 DCD against the guidance provided in NUREG-0800 Chapter 7. With the exception of the items listed above, the staff finds the PMS design descriptions to be acceptable.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

7.3 Engineered Safety Features Actuation Systems

7.3.1.4.1 Automatic Depressurization System Valve Block

The AP1000 design uses ADS valves in certain accident conditions to depressurize the RCS and allow passive safety systems to inject coolant to the reactor core. However, a potential CCF of the PMS could initiate the ADS and simultaneously cause the PMS to not appropriately respond to the condition. As a result, the plant would need to respond to the spurious actuation of the ADS valves using back-up systems, such as the DAS, nonsafety systems, and operator action. While the scenario is beyond design basis, the consequences of such a scenario would be similar to that of up to a large-break LOCA. In addition, the staff did not identify analyses demonstrating the capability of the plant to mitigate the scenario of a spurious actuation of the ADS valves.

To address the potential spurious actuation of ADS valves due to a CCF of the PMS, the applicant added an ADS valve block feature to prevent the spurious actuation of any of the ADS valve paths in each of the four stages of valve sets. The blocking feature prevents one valve in each of the two-valve valve sets from opening without deactivation of the block signal. For ADS Stages 1-3, the blocking feature is applied to the depressurization valve, and regarding ADS Stage 4, the blocking feature is applied to the squib valve. For ADS Stage 4, the applicant will determine whether the “arm” or “fire” signal within the squib valve will have the blocking feature applied. However, based upon the information provided, the staff determined the use of either signal is acceptable.

Section 7.3.1.2.4.1 of the AP1000 DCD describes the need for a confirmatory process condition that is separate from the PMS actuation logic to serve as the input to the blocking signal circuit. The DCD describes the use of redundant CMT level switches, one in each CMT, to serve as the permissive that removes the block signal. Each of the respective level switches act to clear the block signal upon a lowering level in a given CMT once the tank level has reached or surpassed its setpoint. The DCD further states the blocking device will be a Class 1E module that will satisfy the requirements of safety-related I&C equipment. On a divisional basis, the interface between PMS and the blocking device will be the CIM for the each affected ADS valve in the division. The CIM resides in the PMS circuitry after the logic function has taken place. Additionally, the AP1000 DCD states that there are no interdivisional connections between the blocking devices nor will there be any coincidence voting between the blocking devices thus satisfying the independence requirements of IEEE Std. 603-1991. Section 7.3.1.2.4.1 of the AP1000 DCD also discusses the use of manual switches to enable the operator to manually clear the block signal as required. Since this action affects a component with onerous consequences the staff expects, that beyond the commitment in the AP1000 DCD for this switch to reside in the MCR, the given switches will be separate, hardwired switches in the primary dedicated safety panel or another safety-related panel in the MCR. In addition, the applicant updated Figure 7.2-1 noting that an ADS valve block signal is utilized via the CMT level switches’ signal.

Beyond the redundancy created through the use of two CMT level switches to prevent a single level switch from causing a blocking circuit failure as described in the AP1000 DCD, in accordance with the description in the FMEA for the AP1000 PMS, WCAP-16438-P, the text

explains that should an ADS block signal within a division fail to remove the block signal, only a single division of PMS is affected. For Stages 1-3 ADS valves the other division's ADS valves will actuate and as Stage 4 ADS valves utilize signals from two divisions, the "other" PMS division's signal will actuate the given Stage 4 ADS valve.

Based upon a review of the information related to the ADS valve blocking circuit provided in the AP1000 DCD and the AP1000 FMEA for the PMS, the staff finds the addition of the ADS valve block circuit to be acceptable. In a subsequent revision to the AP1000 DCD and the FMEA, the applicant incorporated these changes, which resolves this issue.

7.3.4 ESFAS Evaluation Findings and Conclusions

The ESFAS is a portion of the PMS. AP1000 DCD, Section 7.2 discusses the additional design information associated with the PMS and the staff's evaluation. The staff identified no changes of substance in Section 7.3 of the AP1000 DCD other than those described below.

To prevent spurious actuation in of any of the valve paths for any and all of the stages of the ADS valves, the applicant chose to implement an ADS valve blocking circuit that prevents the given depressurization valve (for Stages 1-3 ADS valves) or the squib valve (for Stage 4 ADS valves) from opening unless system conditions warrant. Based upon a review of the information in AP1000 DCD, Section 7.3.1.2.4.1 and the FMEA for the AP1000 PMS, WCAP-16438-P, the staff finds the addition of the ADS valve block circuit to be acceptable These changes were incorporated in Revision 19 to the AP1000 DCD and WCAP-16438-P, Revision 3, which resolves this issue.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

7.4 Systems Required for Safe Shutdown

7.4.3 Evaluation Findings and Conclusions

The applicant changed the fifth bullet in Section 7.4.3.1.3 of the AP1000 DCD to clarify that the remote shutdown workstation is designed, not for a single failure, but with redundancy. However, when a random event other than fire causes an MCR evacuation, a coincident single failure in the safety systems controlled from the remote shutdown workstation is considered.

The staff finds the change acceptable, since RG 1.189, "Fire Protection for Operating Nuclear Power Plants," establishes the bases for safe shutdown with respect to fire protection. These limits do not require consideration of an additional, random, single failure in the evaluation of the capability to safely shut down because of fire. NUREG-0800 Section 9.5.1 addresses conformance to RG 1.189. Therefore, the application of the single-failure criterion to remote shutdown is applicable only to other events that could cause the control room to become uninhabitable. These events would not result in consequential damage or unavailability of systems required for safe shutdown. The AP1000 design does consider events other than fire, coincident with a single failure in the safety system, for the remote shutdown workstation.

7.5 Safety-Related Display Information

7.5.3 Network Gateway (Real Time to Protection and Safety Monitoring System)

The applicant removed the communications description from Revision 17 of AP1000 DCD Tier 2, Section 7.1.2.8, and added a reference to Section 3 of WCAP-16675-P, Revision 3. WCAP-16675-P, as supplemented by WCAP-16674-P, Revision 2 provides a comprehensive description of the communications within the safety system, between safety and nonsafety systems, and within the nonsafety systems. Section 7.9 documents the review of the modifications made to the AP1000 safety and nonsafety communications system, and provides additional information on the communications system.

7.5.5 Qualified Data Processing System

Revision 17 of the AP1000 DCD upgraded several variables in Table 7.5-1, "Post-Accident Monitoring System," to add seismic qualification to some instruments and to add a QDPS indication. The staff accepts this change, which increases safety with more highly qualified instruments and controls, as well as improved information to support MCR operations.

Revision 17 of the AP1000 DCD changed the information given for several variables in Table 7.5-1 and added Note 7 indicating, "This instrument is not required after 24 hours."

The staff finds the addition of Note 7 acceptable for these variables because AP1000 DCD, Section 7.5.4, includes the statement below:

> Class 1E position indication signals for valves and electrical breakers may be powered by an electrical division with 24-hour battery capacity. This is necessary to make full use of all four Class 1E electrical divisions to enhance fire separation criteria. The power associated with the actuation signal for each of these valves or electrical breakers is provided by an electrical division with 24-hour battery capacity, so there is no need to provide position indication beyond this period. The operator will verify that the valves or electrical breakers have achieved the proper position for long-term stable plant operation before position indication is lost. Once the position indication is lost, there is no need for further monitoring since the operator does not have any remote capability for changing the position of these components.

7.5.5.1 Combined License Information and Tables 7.5-1 and 7.5-8

Section 7.5.5 of the AP1000 DCD states: "This section has no requirement for information to be provided in support of the Combined License (COL) application." Section 7.5, Tables 7.5-1 and 7.5-8, indicate that the meteorological parameters and environs radiation and radioactivity variables are "site specific." The staff requested clarification of this inconsistency in RAI-SRP7.5-ICE-02. The applicant's response states: "The words 'site specific' for environs radiation and radioactivity parameters indicate that these variables are site-related and are to be addressed by the COL applicant in the site emergency response plan identified in DCD Tier 2, Chapter 13, Section 13.3.1. Therefore, each COL applicant is to provide information for monitoring the meteorological parameters and environs radiation and radioactivity as appropriate." In a letter dated May 26, 2010, the applicant provided a revised response to Open Item OI-SRP7.5-ICE-01 stating that the applicant commits to update Tier 2 Tables 1.8-1 and 1.8-2 in Revision 19 to the DCD. The applicant plans to modify Section 7.5.5 to indicate a

COL action item regarding meteorological parameters and environs radiation and radioactivity instrumentation is required by the COL applicant. The staff previously identified this as Open Item OI-SRP7.5-ICE-01 and, based upon the commitment in the applicant's response to the open item, the open item is considered resolved. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

7.5.6 Bypass and Inoperable Status Information

The applicant removed the description of the bypass and partial trip in AP1000 DCD Tier 2, Section 7.1.2.9, and provided a reference to Section 6 of WCAP-16675-P, Revision 5. Section 6.4 of WCAP-16675-P describes the design requirements for the bypass and partial trip conditions.

This section establishes the use of the Common Q network to provide bypass status indication in the MCR, in accordance with RG 1.47.

Evaluation

Regulations in 10 CFR 50.34(f)(2)(v) require the safety-system design to provide an automatic indication of the bypassed and inoperable status of safety systems. As applied to data communications systems, NUREG-0800 Section 7.9 states that the bypassed and inoperable status indications for data communications systems should be consistent with those of the systems of which they are part. The bypassed and inoperable status indication should conform to the guidelines of RG 1.47.

The staff finds the AP1000 DCD, as supplemented by WCAP-16675, sufficiently demonstrated how the PMS design provides bypass indications of protection channels used in the reactor trip and/or ESF actuation path to meet the requirements of 10 CFR 50.34(f)(2)(v). Specifically, the staff concludes the applicant provided adequate information that describes how the indications of bypassed channels or components on the MTP and MCR conform to RG 1.47, as specified in the guidance provided in NUREG-0800 Section 7.9. The staff previously identified this issue as Open Item OI-SRP-7.5-ICE-02, and based upon the information in Section 6.4, "Bypass and Partial Trip Condition," in WCAP-16675, the staff finds the design of the system's ability to inform the operators of bypassed and inoperable safety channels within the PMS to be acceptable.

7.5.7 In-Core Instrumentation System

7.5.7.1 In-Core Instrumentation Interaction with Core Exit Thermocouples

WCAP-17226, "Assessment of Potential Interaction between the Core Exit Thermocouple Signals and the Self-Powered Detector Signals in the AP1000™ In-core Instrumentation System," Revision 2, describes how the AP1000 IIS design satisfies the requirements of IEEE Std. 384-1981 such that any credible single fault in the non-Class 1E self-powered detector (SPD) signals will not reduce the number of valid Class 1E CET inputs to the PAMS below the required minimum number. The CET signal wires used by the Class 1E PAMS and the non-Class 1E SPD signal wires used by the on-line power distribution monitoring system (OPDMS) are in very close proximity, and do not satisfy the separation distances identified in IEEE Std. 384-1981. In addition, the required separation distance identified in IEEE Std. 384-1981 between safety and nonsafety signals is not met in the in-core instrument thimble assemblies (IITAs) and in the mineral insulated (MI) cable assembly hardware that route the

CET and SPD signals from the reactor vessel head (RVH) penetrations to the refueling disconnect panel (RDP). Furthermore, four of the AP1000 IITA contain non-Class 1E CET sensors that provide input to the non-Class 1E DAS.

Within the IITA, the active portions of the Class 1E CET elements and the non-Class 1E SPD elements are placed inside individual steel outer sheaths that share a common ground to provide electrical isolation between the CET and SPD elements. The presence of two commonly grounded metallic barriers within the IITA probe assembly and in the MI cables makes it incredible for an SPD emitter signal to short directly to the CET element signal leads.

From the Quickloc flanges on the RVH to the RDP, the CET signals and SPD signals share a common Class 1E design and post-accident environmentally qualified MI cable assemblies. The MI cable assemblies consist of individual flexible steel-sheathed cable sub-assemblies, which contain the separate CET and SPD signal lead MI cables, routed together in a flexible steel outer sheath that serve as conduits for individual CET and SPD cables in the cable assembly.

From the Class 1E connector panels on the RDP, the SPD signals are split and sent to two signal processing system (SPS) cabinets that are powered by non-Class 1E power supplies. The CET signals are also split at the RDP Class 1E connector panels, where most of the CET signals are sent to the PMS; the remaining 4 CET signals are sent to the DAS. The CET signals sent to the PMS are split into two corresponding trains and sent to the corresponding PMS divisions via separate qualified Class 1E MI cables. The CET signals sent to the DAS are routed through post-accident environmentally qualified MI cables.

An analysis was performed to determine whether the non-Class 1E power supplies that power the SPS cabinets can have an over-voltage or surge voltage that propagate backwards to the SPD inputs signals through the SPS circuitry without attenuation or shorting to ground. The analysis showed that it is credible that a sufficiently large over-voltage or a voltage surge at the SPS cabinet power supply inputs could cause at least a momentary loss of all Class 1E CET signals associated with the affected SPS cabinets via shorting between the SPD and CET wires in the backshell of the IITA or MI cable electrical connectors. If the over-voltage or transient surge condition were to occur on both SPS cabinets, then the result could be that all of the CET signals needed by the PAMS become inoperable.

To mitigate this concern, the applicant performed an analysis to determine the maximum sustained over-voltage value for low voltage circuits in Westinghouse NPP designs that run cable in accordance with nuclear industry standards. The staff requested the applicant describe how the maximum credible over-voltage generated by the SPS cabinets is established. In its response, the applicant proposed to modify WCAP-17226 with the following explanation:

> AP1000 requires that low voltage systems be installed in a separate raceway system from medium voltage systems. As such, the maximum credible sustained overvoltage condition which can occur in a low voltage power or control circuit routed in this (these) low voltage raceway system(s) can be determined conservatively by considering nominal system operating voltages and maximum preferred system voltage range as defined in ANSI C84.1-2006. The system voltage at the low voltage system will remain balanced when the medium voltage system is supplied from normal or reserved source of power during the normal plant operation. During the abnormal plant operation when the normal and reserve sources of power are not available, the low voltage system will

continue to function by receiving power from the standby diesel generators. The system voltage will also remain balanced even when the medium voltage continues to operate in the presence of a single line to ground fault indefinitely.

As the neutral of the load center transformers secondary windings are solidly (or effectively) grounded there will be no increase in the maximum credible sustained overvoltage of the low voltage system whether a ground fault is present at the medium voltage system or not. As such, the maximum credible sustained overvoltage numerical value, when the transformer is lightly loaded, can simply be derived from consideration of the nominal low voltage system of 480VAC plus a 10 percent multiplier (480VAC)(1.10) = 528 VAC for an maximum sustained system voltage during both normal full load and light load operation. The high voltage taps of the load center transformer is set such that the maximum allowable voltage at the terminals of the loads and the secondary winding of the load center transformers is not exceeded.

For purposes of defining a value for design of isolation devices a margin of 10 percent will be used yielding (528VAC)(1.10) = 580VAC. This value is developed specifically for use as a bounding design value for isolation devices and, as described above, is conservative beyond the actual design operating conditions of the plant.

This established maximum credible over-voltage value allows for the identification of operating characteristics of the IITA and MI cable hardware used in the IIS to be specified to withstand a peak over-voltage beyond the identified historical maximum over-voltage value. The design requirements for the MI cable and IITA electrical connector hardware require that manufacturing or proof testing be performed to demonstrate compliance with a 600 volts alternating current (VAC) peak voltage CET functional interaction exclusion requirements. This hardware testing requirement satisfies the requirements for testing or analysis of associated circuit interaction with Class 1E circuits in IEEE Std. 384-1981 for over-voltage conditions. To further mitigate the possibility of a transient surge voltage condition in the SPS cabinet's input power supply in excess of the identified maximum over-voltage value that may disable both divisions of the CET signals used by the PAMS, different divisions of safety power are supplied to the IIS SPS cabinets, with the power cables routed in separate shielded conduits.

The applicant also identified the DAS as another non-Class 1E system that can cause a surge or over-voltage faults to propagate to the IIS. The applicant's analysis found that the maximum credible surge voltage output from the DAS to the DAS CET signal leads that could produce an interaction with the IIS is the same as the IIS IITA and CET cable. The identified maximum credible voltage output from the DAS to the CET signal leads are also equivalent to the electrical connector hardware voltage environmental electromagnetic interference qualification limit requirements in Tier 2, Appendix Section 3D.4.1.2 of the AP1000 DCD. The DCD hardware requirements specifically require that the IIS IITA and associated cables be qualified to meet RG 1.180, "Guidelines for Evaluating Electromagnetic and Radio-Frequency Interference for Safety-Related Instrumentation and Control Systems," peak surge voltage pulse levels. This result ensures that if there is a voltage surge from DAS that propagates down through the DAS CET signal leads to the associated SPD cables, there will be no credible, systematic shorting of DAS CET signals to the associated SPD signal leads, which contain the CET signals that are sent to the PMS.

IEEE Std. 603-1991, Clause 5.6.3, requires that equipment in other systems that is in physical proximity to safety system equipment but that is neither an associated circuit nor another Class 1E circuit be physically separated from the safety system equipment to the degree necessary to retain the safety systems' capability to accomplish their safety functions in the event of the failure of nonsafety equipment. Physical separation may be achieved by physical barriers or acceptable separation distance. The separation of Class 1E equipment shall be in accordance with the requirements of IEEE Std. 384-1981.

The NRC issued RG 1.75, "Criteria for Independence of Electrical Safety Systems" to endorse IEEE Std. 603-384 with exceptions. RG 1.75 identifies the underlying separation criteria: (1) physical separation, and (2) electrical isolation are provided to maintain the independence of safety-related circuits and equipment so that the safety functions required during and following any design-basis event can be accomplished. Section 5.6(3) of IEEE Std. 384-1992 provides general criteria for independence between safety-related and nonsafety-related circuits. When minimum separation cannot be met, it allows an analysis of nonsafety-related circuits to demonstrate that the safety-related circuits are not degraded below an acceptable level. If the analysis is successful, the nonsafety-related circuits can remain as nonsafety-related circuits. RG 1.75 clarifies that: (1) nonsafety-related circuits that are not separated from safety-related circuits through the minimum separation or barriers must be treated as "associated circuits," and (2) the cables that are associated because they are powered from a safety-related source serving nonsafety-related loads or share the safety signal must also be treated as associated circuits. This regulatory guide defines "associated circuits" as "nonsafety-related circuits that are not physically separated or not electrically isolated from safety-related circuits by acceptable separation distance, safety class structures, barriers, or isolation devices."

Based on the staff's evaluation of the information presented in WCAP-17226, the staff finds that the associated circuit analysis performed and the ensuing design requirements adequately ensure that the Class 1E CET elements cannot be degraded below an acceptable level by over-voltage or surges from non-Class 1E SPD elements or other connected nonsafety systems. The specific evaluation of portions of this analysis is documented below:

- Within the IITA, the staff finds that placing the active portions of the Class 1E CET elements and the non-Class 1E SPD elements inside individual steel outer sheaths that share a common ground provides adequate electrical isolation between the CET and SPD elements. The staff finds that since the steel outer sheaths share a common ground, it can adequately protect the CET element from a potential overload of the SPD element.

- From the Quickloc flanges to the RDP, the staff finds the MI cable assembly adequately ensures the isolation of Class 1E CET elements from SPD elements through the use of separate individual flexible steel-sheathed cable subassemblies for the CET and SPD signal leads. The individual steel-sheathed cables provide adequate electrical isolation to prevent faults within the SPD signal leads from propagating to the CET signal leads.

  The staff finds adequate the analysis provided to evaluate the maximum credible over-voltage or surge voltage that can propagate backwards from the non-Class 1E power supplies in the SPS cabinets to the SPD input signals. Specifically, the staff finds that the applicant's analysis, which specifies that, for 3-wire low voltage AP1000 systems, the maximum sustained over-voltage will incorporate a 10 percent margin (which is above the 5 percent margin specified in ANSI C84.1-2006, "Electric Power Systems and Equipment - Voltage Ratings (60 Hz)") for a solidly grounded system is

consistent with the criteria found in IEEE Std. 141-1993, "IEEE Recommended Practice for Electric Power Distribution for Industrial Plants." IEEE Std. 141-1993, Clause 7.2.5, states there are three levels of conductor insulation for medium-voltage cables that are permitted: 100, 133, and 173 percent. The solidly grounded system permits the use of 100 percent insulation level, which indicates that the maximum sustained over-voltage is at 100 percent of the line-to-ground voltage during a single line to ground fault. Thus, the 10 percent additional margin sufficiently bounds the anticipated maximum sustained over-voltage. Ultimately, the cable insulation selection for a nominal low voltage system of 480 VAC will be rated at a minimum of 600 volts (V), which encompasses the anticipated maximum sustained over-voltage value of 528 VAC. The derivation of the maximum credible over-voltage was incorporated in Revision 2 to WCAP-17226.

- The applicant proposed to specify design requirements for the MI cable and IITA electrical connector hardware to be tested to withstand the identified maximum credible voltage values. This approach is acceptable to the staff in that it meets the associated circuit requirements of IEEE Std. 384-1981 for over-voltage conditions. In addition, assigning each SPS cabinet and its corresponding PAMS division to a different Class 1E power bus, the staff finds this approach ensures any fault on the SPS cabinets' safety input power supplies will only occur on one SPS cabinet and can, therefore, only disable one division of the CET signals used by PAMS. The staff finds this acceptable.

- Furthermore, the staff finds adequate the identified maximum credible overvoltage that can be generated from the DAS that could produce an interaction with IIS. The staff finds that by qualifying the IIS IITA and associated cables in accordance with RG 1.180, the design characteristics ensure that a potential voltage surge from the DAS will not be able to cause systematic shortening of the DAS CET signal to SPD signal leads. Therefore, the staff finds that the PAMS CET coverage is adequately protected against failures of the DAS.

7.5.9 Evaluation Findings and Conclusions

Appendix D.II, "Definitions," to 10 CFR Part 52, defines COL action items (COL license information) as items that identify certain matters that applicants must address in the site-specific portion of the final safety analysis report (FSAR). The applicant selected the site-specific parameters required to meet 10 CFR 50.34(f)(2)(xvii), regarding accident monitoring instrumentation as a COL action item and updated Tier 2 Tables 1.8-1 and 1.8-2, accordingly. Therefore, the staff finds the above requirements have been satisfied via the commitment in the applicant's response dated May 26, 2010. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

Regulations in 10 CFR 50.34(f)(2)(v) require the applicant to provide for automatic indication of the bypassed and inoperable status of safety systems. The staff finds that the applicant has sufficiently addressed this requirement in the AP1000 DCD and the supporting TRs, as documented in Section 7.9.2.6. Specifically, the applicant has demonstrated how the indication of bypassed and operable status of safety systems in the MCR conforms to RG 1.47. The staff previously identified this issue as Open Item OI-SRP7.5-ICE-02, and based upon the discussion in Section 6.4 of WCAP-16675, the staff finds the issue has been satisfactorily resolved.

Based upon the staff's review of WCAP-17226-P, the interaction between the IIS and other safety-related systems is deemed acceptable provided the proposed change to WCAP-17226-P to include a discussion of how the maximum credible over-voltage value is derived is included in

Revision 2 of the document. These changes were incorporated in Revision 2 to WCAP-17226-P, which resolves this issue.

Additionally, the staff finds that the accident monitoring instrumentation meets the applicable requirements of GDC 13 and GDC 19, "Control Room."

7.6 Interlock Systems Important to Safety

7.6.5 Evaluation Findings and Conclusions

Section 7.3.1.4.1 of this report describes the applicant's commitment to add an ADS valve blocking circuit to prevent the spurious actuation of any of the valve paths in any of the stages for the ADS. As this circuit is described as a blocking circuit rather than an interlock (in which a block may be cleared or overridden but an interlock must not), the circuit's operation is not captured in Section 7.6 of this report. As such, the changes in Section 7.6 of the AP1000 DCD did not affect any conclusions regarding regulatory compliance in NUREG-1793. Therefore, the staff finds the applicant continues to meet the requirements identified in NUREG-1793 for Section 7.6.

7.7 Control and Instrumentation Systems

7.7.1 System Description

7.7.1.1 Reactor Power Control System

The applicant revised the second paragraph of Section 7.7.1.1.1, "Power Control," to remove the term "lead/lag compensated." The applicant revised the AP1000 design to apply the lead/lag compensation after the high auctioneer, versus before it. The applicant claims that this design does not require compensation to be factored into signal quality check acceptance criteria in the auctioneer. The applicant described this justification in APP-GW-GLR-080, Revision 0, "Mark-up of AP1000 Design Control Document Chapter 7." The staff's review confirms that this design change does not affect any staff conclusions in NUREG-1793.

The applicant also revised the description of the T_{ave} reactor control band for the various plant modes of control (i.e., load follow, load regulation, base load). The applicant claims that there is no advantage to increasing the deadband during load follow operations. The applicant further states that doing so would erode margins to reactor trip setpoints. The applicant also stated this justification in APP-GW-GLR-080, Revision 0. The staff's review confirms that this change does not affect any staff conclusions in NUREG-1793.

The applicant removed the "time weighted average" or "smoothing" compensation to the nuclear flux and the axial offset signals while the plant is in a load regulation mode of control. The applicant states that this current transient specification does not require complex "time weighted average" nuclear flux and axial offset signal compensation on the inputs to the axial offset control band calculation and that simple lag compensation is adequate. The applicant also stated this justification in APP-GW-GLR-080, Revision 0. The staff's review confirms that this change does not affect any staff conclusions in NUREG-1793.

7.7.1.2 Rod Control System

The applicant revised the interlock and "low" and "low-low" alarms associated with control rod insertion limits. The revision will move the automatic activation of the rod insertion interlock from the "low" rod insertion alarm setpoint to the "low-low" rod insertion alarm setpoint. In moving the activation of the rod insertion interlock from the "low" to the "low-low" rod insertion alarm, rod insertion will be prevented, at the "low" alarm setpoint, by following appropriate plant operating procedures and will be prevented at the "low-low" setpoint by automatically actuating the rod insertion interlock and terminating automatic AO bank insertion (or withdrawal to prevent further M bank insertion).

By moving the rod insertion interlock to the "low-low" alarm setpoint, continued rod insertion (and thus, a continued reduction in control rod shutdown margin) is automatically terminated by plant controls versus "appropriate plant operating procedures."

Furthermore, removing the interlock from the "low" alarm setpoint does not reduce any plant protection function or increase the risk of reducing protection against a reduction in control rod shutdown margin caused by the margin built into the difference between the "low" and the "low-low" rod insertion setpoints. Therefore, the staff's review confirms that this design change does not affect any staff conclusions in NUREG-1793, including Supplement 1.

7.7.1.3 Pressurizer Pressure Control

The applicant changed the description of the pressurizer variable heating control by stating that it is not sensitive to the rate of change in pressure and that it will respond in the same manner to small, fast, or slow small changes in pressure. The staff's review confirms this change does not affect any staff conclusions in NUREG-1793.

7.7.1.5 Feedwater Control

In Section 7.7.8.1 of the AP1000 DCD, the low-range feedwater flow measurement is no longer used in the low-power mode, and it is not used in the integration (reset) action of the low-power mode feedwater flow controller. This means that only feedwater temperature (low-power mode) and steam flow (high-power mode) are used to tune the integrator setpoints.

In both Sections 7.7.8.1 and 7.7.8.2, the control of the lift on the main and startup feedwater control valves is no longer determined by the ΔP available across the feedwater control valve, and the flow coefficient (C_v) characteristic of the valve. Therefore, in high-power control mode, the feedwater flow is regulated in response to changes in steam flow and proportional plus integral (PI)-compensated steam generator narrow-range water level deviation from setpoint. In the low-power control mode, the feedwater flow is regulated in response to changes in steam generator wide-range water level and PI-compensated steam generator narrow-range water level deviation from setpoint.

The startup feedwater control subsystem regulates the flow of feedwater in a manner similar to the way the (main) feedwater is controlled in the low-power control mode. Feedwater flow is regulated in response to changes in the steam generator wide-range water level and PI-compensated steam generator narrow-range water level deviation from setpoint. The staff's review confirms that this design change does not affect any staff conclusions in NUREG-1793.

7.7.2 Diverse Actuation System

Section 7.8 includes a detailed evaluation regarding the DAS design changes.

7.7.3 Signal Selector Algorithms

The applicant has not demonstrated what specific actions are taken by the signal selector algorithms (SSAs) in the event that one of the multidivisional or multichannel inputs is deemed faulty or of "bad" quality. The staff requires all outputs of the device, whether they are in the form of control, alarm, interlock, or indications, to be identified and addressed. The staff issued this request to the applicant as RAI-SRP7.7-ICE-01. The applicant responded to the RAI in a letter, dated July 7, 2008. However, the staff found the response inadequate. The applicant submitted Revision 1 to address this issue on May 6, 2009, and subsequently the applicant submitted Revision 2 of RAI-SRP7.7-ICE01. In the Revision 2 response, submitted on January 27, 2010, the applicant states it will update the forthcoming revision of the AP1000 DCD so that it is clear that the SSAs are executed in the PLS and additionally that PMS and DAS performance are independent of the SSA. The staff previously identified this as Open Item OI-SRP7.7-ICE-01. Revision 2 of RAI-SRP7.7-ICE-01 adequately addresses this issue. In a subsequent revision to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

7.7.4 Evaluation Findings and Conclusions

The staff finds the conclusions described in NUREG-1793 still valid, based on the staff's review of the changes proposed in the AP1000 DCD. Specifically, the staff required additional information on how the SSAs would affect the PMS and the DAS. Information on such impacts could affect the degree of independence between the control system and the protection system, as required in GDC 24. Also, such impacts could affect the degree of diversity and quality of the DAS as required in GDC 22. The response discussed in Section 7.7.3 above, adequately addressed the issue. The staff previously identified this as Open Item OI-SRP7.7-ICE-01.

Section 7.8 includes further evaluation of the DAS.

7.8 Diverse Instrumentation and Control Systems

7.8.1 System Description

The I&C systems reviewed in this section include the diverse I&C systems and equipment that provide a diverse backup to the safety-related PMS and the defense against postulated CCFs in the PMS and the nonsafety-related PLS concurrent with postulated transients.

The review ensured that the applicant designed and installed the anticipated transient without scram mitigation systems and equipment in accordance with the requirements of 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients without Scram (ATWS) Events for Light-Water-Cooled Nuclear Power Plants," and 10 CFR Part 50, Appendix A, GDC 22.

The DAS in the AP1000 DCD is a nonsafety related I&C system and provides a diverse and independent method for tripping the reactor and performing several ESFs in order to meet the requirements of 10 CFR 50.62 and 10 CFR Part 50, Appendix A, GDC 22. In addition, a set of dedicated and independent displays of selected plant indications and manual controls is

provided in the MCR to address the criteria in BTP 7-19, "Guidance for Evaluation of Diversity and Defense-in-Depth in Digital Computer-Based Instrumentation and Control Systems."

The scope of the safety evaluation in the following section is limited to the changes that have been made to the approved DAS system since the AP1000 DCD, Revision 15, was certified.

7.8.2 Diverse Actuation System Assessment

The findings in NUREG-1793, Supplement 0, related to DAS functionality are based upon WCAP-15775, "AP1000 Instrumentation and Control Defense-in-Depth and Diversity Report," (APP-GW-J1R-004), Revision 0, issued April 2002. The applicant submitted Revision 2 of its D3 assessment on April 3, 2003. The applicant revised its D3 assessment based on the response to the NRC's RAI 420.013, Revision 1, in a letter dated February 21, 2003. The applicant referred to the Revision 2 of the D3 assessment in the certified AP1000 DCD, Revision 15. In the certified AP1000 design the hardware of the DAS is microprocessor based and its operating systems or programming languages are different from those used in the PMS.

In the original NUREG-1793, Section 7.1.6, the staff concluded that "the proposed design satisfies the Commission's position on I&C system diversity." The NRC based its conclusion on the proposed DAS design, as stated in Revision 15 of the applicant's DCD, which included the ITAAC listed in AP1000 DCD Tier 1, Section 2.5.1.

In the second paragraph of Section 7.7.2 of NUREG-1793, the staff concluded that based on the design commitments of Revision 15 of the AP1000 DCD:

> The DAS automatic actuation is accomplished by a microprocessor-based system. Diversity from the PMS is achieved by using a different architecture, different hardware implementation, and different software. Software diversity is achieved by running different operating systems and programming in a different language.

The applicant submitted TR-97, Revision 1, in which it changed the DAS design commitments and DAS ITAAC in Revision 15 of the AP1000 DCD to allow non-microprocessor-based implementations, added the DAS instrumentation cabinet, added an electrical penetration to the containment, and relocated a portion of the DAS to another area of the plant.

The following evaluation focuses on the modification to allow non-microprocessor-based implementations of the DAS and the addition of the DAS instrumentation cabinet. The relocation of DAS equipment within the plant has no impact upon the I&C review. Additionally, the installation of another electrical penetration to the containment is beyond the scope of the Chapter 7 review.

For the purposes of the Chapter 7 I&C review, the following areas of the DCD are affected:

Tier 1
Section 2.5.1
Table 2.5.1-4
Table 2.5.1-5
Table 3.7-1
Tier 2
Section 7.7.1.11
Figure 7.2-1
Section 9A.3.1.3.1.1
Table 14.3-3
Table 14.3-6
Table 17.4-1
Table 19.59-18

The applicant discussed the change to the I&C technology utilized for the DAS throughout Tier 1 and Tier 2 in Revision 16 of the AP1000 DCD by substituting language, as appropriate, where it used the terms "microprocessor" and "software" to describe DAS technology. The applicant replaced the term "microprocessor" with "microprocessor or special purpose logic processor," and the term "software" with "any software." The addition of the DAS instrument cabinet, DAS-JD-004 will contain the 4-20 mAdc loop transmitters and power supplies necessary to complete the DAS instrumentation requirements. The DAS-JD-004 cabinet mounts next to the DAS squib valve control cabinet, DAS-JD-003. This is acceptable. The applicant revised Section 7.7.1.11 of the AP1000 DCD Tier 2, Revision 16, by replacing the terms "microprocessor-based" and "microprocessor" with "logic", "software" with "any software" from several design change descriptions in APP-GW-GLR-080, Revision 0, which describes all changes to the AP1000 I&C systems in the certified AP1000 DCD.

This proposed change to use non-microprocessor-based technology is intended to increase reliability and was found to be acceptable. However, a discussion concerning the diversity between the PMS and the DAS is provided later in this section.

In Revision 17 of the AP1000 DCD, the applicant removed DAS DAC from Tier 1, Section 2.5.1, "Diverse Actuation System (DAS) Design Description," Items 4a and 4b, as well as from Tier 1, Table 2.5.1-4, ITAAC Items 4a and 4b. Items 4a and 4b are the design requirements and system definition phases of the DAS hardware and software design process. The applicant removed portions of the DAC, and also provided the corresponding design information in WCAP-17184-P, Revision 2, which addresses those two phases. However, the applicant failed to provide a description in Chapter 7 of the AP1000 DCD, Tier 2, regarding completion of those two phases found in the AP1000 DCD Tier 1. 10 CFR 52.47(a)(2) requires standard DC applications to provide a level of design information sufficient to enable the Commission to reach a final conclusion on all safety issues associated with the design before the certification is granted. In the proposed amendment to the AP1000 DCD, Tier 1, ITAAC Items 4a and 4b are removed based on design work accomplished. In Chapter 7 of the AP1000 DCD, Tier 2, the applicant should provide a summary and justification for why the ITAAC found in Items 4a and 4b of AP1000 DCD, Tier 1, Table 2.5.1-4, can be removed. The staff issued RAI-SRP7.8-DAS-12, requesting the applicant describe the completeness of the above two phases in Chapter 7 of the AP1000 DCD, Tier 2. In response to this RAI, the applicant provided a detailed description about the two completed life cycle phases as a markup for AP1000 DCD Tier 2, Section 7.7.1.11, which is found acceptable. In addition, the applicant added an ITAAC item to Table 2.5.1-4 of the AP1000 DCD Tier 1 to address the ITAAC for DAS manual actuations. The staff finds the above changes to the AP1000 DCD Tier 1 and Tier 2 acceptable

and finds the response to RAI-SRP7.8-DAS-12 acceptable. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

In the initial submittal of WCAP-17184-P, Revision 0, the staff found that the applicant failed to address the DAS setpoint methodology. GDC 13 requires, among other things, that appropriate controls be provided to maintain variables and systems within prescribed operating ranges. The guidance in BTP 7-19, Section B, Item 3, Positions 1 and 2, acceptance criteria address confirmation that an anticipated operational occurrence (AOO) and postulated accidents are mitigated in the presence of CCF.

In WCAP-17184-P, Revision 0, the applicant did not identify how the DAS actuation setpoints and timing delays would be established. DAS must be able to perform its functions to ensure the potential release of radioactive material for postulated accidents and AOOs fall within acceptable limits in the event of a software CCF of the safety-related protection system. The staff issued RAI-SRP7.8-DAS-02, requesting that the applicant describe the DAS setpoint methodology. To address this RAI, the applicant provided WCAP-17184-P, Revision 1, to add the DAS setpoint methodology description as a new appendix to this TR. After reviewing the DAS setpoint methodology description, the staff found that the DAS allowances for automatic actuation signals, "Containment Temperature High" and "Pressurizer Water Level Low" are outside of the typical ratio of $1.15\sigma/2\sigma$. The staff issued RAI-SRP7.8-DAS-10, requesting that the applicant provide the design basis to support the deviation. In response to this RAI, the applicant proposed a revision to WCAP-17184-P in which it described why the DAS channel statistical allowances for "Containment Temperature High" and "Pressurizer Water Level Low" do not conform to the typical ratio of $1.15\sigma/2\sigma$ when comparing a 75 percent probability/75 percent confidence level to 95 percent probability/95 percent confidence level for determination of the random and independent terms of the square-root-sum-of-the-squares calculation. Additionally, the applicant also provided a justification for the use of a 75 percent probability/75 percent confidence level. After reviewing the revised report, the staff found that the applicant adequately addressed the DAS setpoint methodology and found it acceptable. Therefore, the staff finds the responses to RAI-SRP7.8-DAS-02 and RAI-SRP7.8-DAS-10 acceptable. These changes were incorporated into Revision 2 to WCAP-17184-P.

In the proposed changes to the AP1000 DCD, the applicant changed the microprocessor-based implementation of the DAS to be a special purpose logic processor-based system. This special purpose logic processor-based DAS is further described as an FPGA digital platform-based system in WCAP-17184-P. The applicant also made changes to use the FPGA technology for the CIM in the safety-related PMS in WCAP-17179-P and WCAP-16675-P. According to the above reports, the CIM and DAS systems will be designed and manufactured by the same company at a common design and manufacturing facility. 10 CFR Part 50, Appendix A, GDC 22, requires, among other things, that design techniques such as functional diversity or diversity in component design and principles of operation shall be used to the extent practical to prevent loss of the protection function. BTP 7-19 provides guidance for evaluating an applicant's D3 assessment to ensure conformance with the NRC position on D3 for digital I&C systems. NUREG/CR-6303, "Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems" provides diversity analysis methods and strategies to demonstrate that adequate and sufficient diversity should be included within the design. The staff found that the applicant had not provided design descriptions capable of demonstrating adequate and sufficient diversity between the DAS and CIM systems in accordance with the guidance listed above. Hence, the staff issued RAI-SRP7.8-DAS-04, requesting that the applicant describe in detail how the DAS equipment (i.e., hardware, software) would be diverse from the safety-related CIM in PMS. The staff also issued another RAI-SRP7.8-DAS-05

requiring the applicant to identify the criteria, practices, and processes that will ensure adequate diversity in the development of the CIM and the DAS at the common design and manufacturing facility, including the diversity with respect to human, software, and equipment diversity.

In response to the above RAIs, the applicant states, in part, that diversity is a principle in instrumentation of sensing different variables, using different technology, using different logic or algorithm, or using different actuation means to provide different ways of responding to postulated plant conditions. The applicant also revised WCAP-15775 to Revision 4 to address the specific requirement of diversity between CIM and DAS. The applicant demonstrated in Section 2.11 of WCAP-17179-P, Revision 2, and Section 9 of WCAP-17184-P, Revision 2, how the requirements of diversity are met between CIM and DAS. For example, for the human diversity, the applicant states in the two TRs that different designers are used for the CIM and DAS designs. In addition, the different design teams and different test teams will be used to test the CIM and DAS designs. In order to achieve the software diversity between the DAS and PMS (i.e., CIM), the applicant will use different algorithms, logic, program architecture, executable operating system and executable software/logic. The staff concludes that the applicant has provided sufficient information demonstrating conformance with regulatory policies and criteria concerning diversity. The AP1000 DCD Tier 1 and Tier 2 will also be updated accordingly to address the software diversity. Therefore, the staff finds the responses to RAI-SRP7.8-DAS-04 and RAI-SRP7.8-DAS-05 acceptable.

The staff found inconsistencies between Chapter 7 of the AP1000 DCD, Tier 2, Revision 17 and WCAP-17184-P, Revision 1, regarding DAS manual actuations. 10 CFR 52.47(a)(2) requires that an application must include a sufficient description and analysis of the structures, systems, and components (SSCs) of the facility, with emphasis upon performance requirements, the bases, with technical justificationupon which these requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to the safety evaluation. The design information provided for the design basis items, taken alone and in combination, should have one and only one interpretation. Therefore, the staff issued RAI-SRP7.8-DAS-06 to request the applicant provide correct and unambiguous design descriptions for the list of DAS manual actuations. The applicant clarified that the list in the AP1000 DCD is all-inclusive of any manual actuation performed by DAS. The tripping of the RCPs is only done in conjunction with the CMT actuation and, therefore, not separately listed in the DCD. The list in Section 2.3 of WCAP-17184-P, Revision 1, is the list of manual actuations that are cited in the probabilistic risk assessment (PRA) for enabling AP1000 to meet its large release frequency (LRF) goal for beyond design basis events. Therefore, this list would be different from the list in the DCD because not all of the DAS manual actuations are needed to meet the LRF goal for beyond design basis events. Table B-1 in WCAP-17184-P is also not intended to be a complete list of all of the manual actuations. This table is modified to only include those manual actuations that do not have automatic DAS actuations. To incorporate the above clarification, the applicant proposed a revision to WCAP-17184-P, which the staff found acceptable. Therefore, the staff finds the response to RAI-SRP7.8-DAS-06 acceptable. These changes were incorporated into Revision 2 to WCAP-17184-P, which resolves this issue.

The staff found Appendix B, Table B-1, in WCAP-17184-P, Revision 2, includes the manual actuation of the hydrogen control system or igniters, but it is not credited in Section 2.3 of WCAP-17184-P for the DAS manual operator action and/or a DAS automatic function. The applicant did not provide a technical basis for this non-credited DAS manual operator action. BTP 7-19 states where operator action is cited as the diverse means for response to an event, the applicant should demonstrate that adequate information (indication) and sufficient time is

available for operator action. The staff's review of Section 10.2.1.1 in WCAP-17184-P, Revision 2, determined that the stated design descriptions do not provide an explanation of how the manual action is used in the DAS design beyond listing the manual actions. From the PRA evaluation the staff found that there is a 19-minute window for accomplishing this action. The applicant failed to provide a clear technical basis that will permit sufficient understanding of credited DAS manual actuations and their conformance with the applicable regulatory requirements. The staff issued RAI-SRP7.8-DAS-07 to request the applicant provide the technical basis for the hydrogen igniter manual action as a non-credited DAS function.

In response, the applicant states, in part, that for this manual actuation of the hydrogen control system or igniters, PRA analysis techniques show acceptable results even if the act of manually actuating the hydrogen igniters is not accomplished (operator fails to act with 100 percent certainty). For this reason, manual actuation of the hydrogen igniters is not a credited manual operator action nor is it required (or credited) for hydrogen igniters to operate automatically. The 19-minute window as described in the PRA analysis is for a beyond design basis event. The PRA analysis was used to provide insights into this particular scenario. Hydrogen igniters were added to the AP1000 design even though they are not credited in the design basis. In response to this RAI, the applicant revised WCAP-17184-P to provide the reasoning behind installation of hydrogen igniters. After evaluating the response, the staff found the applicant's response to this RAI acceptable. Therefore, the staff finds the response to RAI-SRP7.8-DAS-07 acceptable. These changes were incorporated into Revision 2 to WCAP-17184P, which resolves this issue.

In Appendix B, Table B-1, in WCAP-17184-P, Revision 1, the staff found that there is a 20-minute window for accomplishing the manual actuation of the ADS. However, after reviewing the above TR, the staff found that the stated design descriptions do not provide an explanation of how the ADS manual actions are used in the DAS design beyond listing them. According to BTP 7-19, the applicant should demonstrate that adequate information (indication) and sufficient time is available for manual operator actions. The staff issued RAI-SRP7.8-DAS-08 to request the applicant provide a clear technical basis description in the TR that permits sufficient understanding of ADS manual operator action as credited DAS manual actuations and the basis for why the 20-minute window is sufficient for completing the ADS manual actuation. In response to this RAI, the applicant revised WCAP-17184-P to provide clarification for this manual actuation of ADS.

The DAS credits manual actions to depressurize the RCS during AOOs or postulated accidents following a software CCF in the protection system. BTP HICB-19, "Guidance for Evaluation of Defense-in-Depth and Diversity in Digital Computer-Based Instrumentation and Control Systems," Revision 4, states that where operator action is cited as the diverse means for response to an event, the applicant should demonstrate that adequate information (indication) and sufficient time is available for operator action. The staff reviews acceptability of these manual actions using guidance in DI&C-ISG-05, "Highly-integrated Control Rooms – Human Factors Issues" (ISG-5).

Manual actions that can be initiated from DAS are listed in the following places:

- AP1000 DCD, Revision 19, Tier 2, Section 7.7.1.11
- WCAP-17184-P, Revision 1, Section 2.3
- WCAP-17184-P, Revision 1, Appendix B, Table B-1

The manual actions listed in WCAP-17184-P, Section 2.3, were not consistent with the other two lists. The staff initiated RAI-SRP7.8-DAS-06 to resolve the differences. In its response of June 22, 2010, the applicant explained that the DC described all manual actions that can be performed by DAS. WCAP-17184-P, Section 2.3, described the manual actions that are cited in the PRA for enabling AP1000 to meet its LRF goal for beyond design basis events. The lists are different because not all of the DAS manual actions are needed to meet the LRF goal for beyond design basis events. Appendix B, Table B-1, provided a list matching the DCD. The applicant will modify Table B-1 to include only those manual actions that do not have corresponding automatic DAS actuations. For these manual actions, the applicant provided specific information describing whether the manual action is credited as part of the DAS response to a CCF or whether the manual action is part of the defense-in-depth strategy for severe accident management, thus providing a clearer communication of why the manual actions are included in the DAS design. These changes were provided in Revision 2 of WCAP-17184-P. The staff found the changes acceptable. These changes were incorporated in a Revision 2 to WCAP-17184-P, which resolves this issue.

Appendix B of WCAP-17184-P now lists the following 4 manual actions:

Manual Action 1: Manual Initiation of IRWST Recirculation/IRWST Drain for In Vessel Retention Support

The DAS provides this capability as part of the defense-in-depth strategy for severe accident management. It is an action needed to address AOOs or postulated accidents following CCFs in the protection systems. Therefore, the regulatory guidance in BTP 7-19 and ISG-05 apply.

Manual Action 2: Manual Initiation of the Hydrogen Control System

The DAS cabinet presents a convenient and reliable location for the manual hydrogen control system switches because it is in the MCR and has a diverse battery-backed power supply. It is not an action needed to address AOOs or postulated accidents following CCFs in the protection systems. Therefore, the regulatory guidance in BTP 7-19 and ISG-05 does not apply.

Manual Action 3: Manual Depressurization of the RCS

This action is credited in the DAS response to a CCF. In summary, ISG-05 states that an analysis must be completed that demonstrates:

- The time available to perform the required manual actions is greater than the time required for the operator(s) to perform the actions.

- The operator(s) can perform the actions correctly and reliably in the time available

No information was provided to the staff to explain how this guidance was addressed. The staff initiated RAI-SRP7.8-DAS-8 requesting this information. In its response, dated June 22, 2010, the applicant committed to include a new ITAAC in Tier 1 Chapter 2, Section 2.5.1, "Diverse Actuation System," Table 2.5.1-4. This commitment was provided in the response to RAI-SRP7.8-DAS-12 and reads as follows:

Table 2.5.1-4. Inspections, Tests, Analyses, and Acceptance Criteria

Design Commitment	Inspections, Tests, Analyses	Acceptance Criteria
5. The DAS manual actuation of ADS, IRWST injection, and containment recirculation can be executed correctly and reliably.	An evaluation to confirm that the operator actions can be performed within the specified times.	b) A report exists and concludes that DAS manual operator action verification was conducted.

The staff finds the proposed ITAAC to be an acceptable way to develop the information needed to address the regulatory guidance in ISG-05, provided the commitment is also included as a separate "Design Description" in the text-based portion of Section 2.5.1. In a subsequent revision of the AP1000 DCD, the applicant added Design Description 5 and Design Commitment 5 to Tier 1 Section 2.5.1 and Table 2.5.1-4 respectively, which resolves this issue.

The staff notes that APP-GW-GL-011, "AP1000 Identification of Critical Human Actions and Risk Important Tasks," Revision 0, Tables 3-1 and 3-2 identify risk-important human actions. Table 3-2, Basic Event ID: AND-MAN01 identifies failure to actuate the ADS for RCS depressurization as recovery from failure of automatic actuation or for manual ADS actuation as a risk-important human action. As such this manual action is already included in the HFE program described in Chapter 18 of this report. In summary, that program includes all risk important human actions as priority items in the task analysis, the human-system interface (HSI) design, the HFE design V&V, and human performance monitoring elements. By being included in the HFE program the regulatory guidance provided in ISG-05 is either met or exceeded.

Manual Action 4: Manual Initiation of IRWST Gravity Injection.

The DAS provides this capability as part of the defense-in-depth strategy for severe accident management. It is an action needed to address AOOs or postulated accidents following CCFs in the protection systems. Therefore, the regulatory guidance in BTP 7-19 and ISG-05 applies.

Operator manual actions credited within the DAS design have not yet been evaluated to verify they are viable. The applicant provided an acceptable DAS ITAAC in Section 2.5.1 of Tier 1, Chapter 2 information that will track completion of this evaluation. A similar evaluation is required within the HFE program described in DCD Chapter 18. Together or independently these commitments will ensure the regulatory guidance in BTP 7-19 and ISG-05 is implemented. This, in turn, will provide reasonable assurance that the operator manual actions to depressurize the RCS are an effective element of the DAS design for coping with AOOs or postulated accidents following CCFs in the protection systems.

Additionally, by the applicant adding a new ITAAC item to Table 2.5.1-4 of the AP1000 DCD Tier 1 to address the design commitment and ITAAC for DAS manual actuations, the staff found the applicant's response to this RAI acceptable. Therefore, the staff considers RAI-SRP7.8-DAS-08 closed. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

The staff found in Appendix B, Table B-1, in WCAP-17184-P, Revision 1 that the applicant includes IRWST gravity injection as an DAS manual actuation. From the PRA evaluation in the table, there is a 20-minute window for accomplishing this manual action. However, the applicant failed to provide the technical basis for the sufficiency of this manual operator action. The applicant should demonstrate that adequate information and sufficient time is available for manual operator actions based on the guidance in BTP 7-19. Therefore, the staff issued RAI-SRP7.8-DAS-09 to request the applicant identify a clear technical basis description that permits sufficient understanding of this credited DAS manual actuation and its basis for why the 20-minute window is sufficient for this DAS manual action. In response to this RAI, the applicant states, in part, that for this particular scenario, PRA analysis techniques show acceptable results even if the act of manually actuating IRWST gravity injection is not accomplished (operator fails to act with 100 percent certainty). For this reason, manual actuation of IRWST gravity injection is not a credited manual operator action nor is it required (or credited) for automatic operation. The 20-minute window as described in the PRA analysis is for a beyond design basis event. The PRA analysis was used to provide insights into this particular scenario. The capability to manually actuate IRWST gravity injection from DAS was added to the AP1000 design out of caution, even though it is not credited in the design basis. The applicant proposed to revise WCAP-17184-P to Revision 2 to provide further clarification of the reasoning behind the manual initiation of IRWST gravity injection. The applicant committed to add a new ITAAC item to Table 2.5.1-4 of AP1000 DCD Tier 1 to address the design commitment and ITAAC for DAS manual actuations, which include the manual actuation of the IRWST gravity injection. The staff found that the applicant's response and changes related to this RAI are acceptable. The staff finds the response to RAI-SRP7.8-DAS-09 acceptable. These changes were incorporated into Revision 2 of WCAP-17184-P, which resolves this issue.

In WCAP-17184-P, Revision 1, which the applicant submitted to justify the removal of ITAAC Items 4a and 4b from AP1000 DCD Tier 1, Table 2.5.1-4, the staff found that the applicant addressed the requirements of cyber security. The requirements of cyber security were also addressed in WCAP-17184-P, Revision 1. However, the staff's position related to cyber security issues is that cyber security is addressed by 10 CFR 73.54, "Protection of digital computer and communication systems and networks," and is not a 10 CFR Part 50 review item. As such, the reference to cyber security in the above TRs should be modified and/or replaced with a docketed TR describing the SDOE in which the applicant chooses to develop its software-based and programmable technology based DAS and CIM, paying particular attention to IEEE Std. 603-1991, Clauses 5.3, 5.6.3, and 5.9. The staff issued RAI-SRP7.8-DAS-11, which requested that the applicant remove the discussion of cyber security from the two TRs. In response to this RAI, the applicant deleted the mention of cyber security requirements in the revised WCAP-17184-P and WCAP-17179-P. The applicant also made corresponding changes to AP1000 DCD Tier 2, Table 1.6-1 and Section 7.1 to address the requirements of this RAI. The staff finds the response to RAI-SRP7.8-DAS-11 acceptable. In addition, the applicant submitted APP-GW-J0R-012, Revision 1, to address the secure development and operational environment for the AP1000 PMS. Section 7.9 provides the evaluation of this new TR.

The staff found that the following issues in WCAP-17184-P, Revision 1, need to be addressed by the applicant:

- Section 6.1.2.2 references the Wolf Creek license amendment request regarding self-test features of the DAS. However, the Wolf Creek license amendment request is not part of the AP1000 DCD licensing basis. Therefore, self-test features should be identified in the TR for the AP1000 DAS.

- The statement in Section 8.1 currently states that the "two-out-of-two logic ... lends itself to reliability." However, this configuration is less reliable than a single train configuration, although the PRA-based DAS design and two-out-of-two (2oo2) actuation logic were approved in the certified AP1000 DCD, Revision 15, and the applicant has not made changes to the DAS logic in the applicant's DC application. Hence, the above statement is not accurate and should be modified to reflect the approved DAS design feature.

- Section 8.1 states that "The use of FPGAs results in a hardware-based design that is not subject to software CCFs. The only software involved in the process is that used to burn-in the required logic design into the FPGA." These two statements are inaccurate and should be removed because the FPGA-based systems are developed with software tools and can have programming errors similar to microprocessor-based digital systems, although FPGA-based systems do not run any system or application software during operation. Therefore, evaluation for CCFs in the FPGA development process and software programming shall be conducted. A high quality and well documented life-cycle design process should be provided according to BTP 7-14, "Guidance on Software Reviews for Digital Computer-Based Instrumentation and Control Systems."

During the review of WCAP-17184-P, Revision 1, the staff identified the above issues and issued RAI-SRP7.8-DAS-13 to request that the applicant correct those statements for technical and regulatory accuracy. In response to this RAI, the applicant proposed to modify WCAP-17184-P, Revision 2, to make the necessary corrections. After reviewing the draft revised TR, the staff found the applicant's response acceptable. The staff finds the response to RAI-SRP7.8-DAS-13 acceptable. These changes were incorporated into Revision 2 of WCAP-17184-P.

The staff finds the DAS is properly credited for providing a diverse backup to the safety-related protection system; however, the ATWS mitigation systems actuation circuitry (AMSAC) system should not have been credited with providing diverse protection upon a postulated CCF of the safety-related PMS. During the evaluation of the changes made to the certified AP1000 DCD, Revision 15 for the DAS, the staff found that the applicant provided ambiguous descriptions for the DAS circuitry and the AMSAC. 10 CFR 52.47(a)(2) requires, in part, a description of the SSCs to be of sufficient detail to permit understanding of the systems designs. Section 15.8 of the AP1000 DCD, states that the DAS provides AMSAC functions. It also states that for Westinghouse plants the ATWS rule (10 CFR 50.62) requires the installation of AMSAC, which consists of circuitry separate from the reactor protection system to trip the turbine and initiate decay heat removal. The applicant failed to provide a description of the AMSAC or the relation between the DAS and the AMSAC system in Section 7.7 of the AP1000 DCD Tier 2. The applicant also failed to clearly describe if the DAS circuitry and the AMSAC system circuitry are the same system or if they are separate systems. The staff issued RAI-SRP7.8-DAS-01 to request the applicant clarify the design descriptions for DAS and AMSAC. In response, the applicant states that for Westinghouse plants the ATWS rule requires the installation of equipment that is diverse from the reactor protection system to automatically trip the turbine and initiate decay heat removal. This equipment must be designed to perform its function in a reliable manner and be independent from sensor output to final actuation device from the existing reactor protection system. The AP1000 is designed with a DAS, which provides the functions required by the ATWS rule. In response to this RAI, the applicant also provided a markup for Section 15.8 of the AP1000 DCD to clarify the description of DAS and AMSAC. The staff found the response to RAI-SRP7.8-DAS-01 acceptable. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

When evaluating the changes to the certified AP1000 DCD, Revision 15, the staff issued RAI-SRP-7.8-DAS-03 requesting the applicant identify design descriptions that demonstrate how the 2oo2 DAS actuation logic would meet the applicable regulatory criteria. 10 CFR 50.62 requires ATWS mitigation equipment to perform its functions in a reliable manner. The guidance of BTP 7-19, Point 3, on D3 states that the diverse or different function may be performed by a nonsafety system if the system is of sufficient quality to perform the necessary function under the associated event conditions. The applicant stated in Revision 17 of AP1000 DCD, Tier 2, Section 16.3-2 that "when a required channel is unavailable, the automatic DAS function is unavailable."

In its response, the applicant explained that there are two actuation logic modes, automatic and manual. The automatic actuation logic mode functions to logically combine the automatic signals from the two redundant automatic subsystems in a 2oo2 basis. The combined signal operates a power switch with an output drive capability that is compatible, in voltage and current capacity, with the requirements of the final actuation devices. The 2oo2 logic is implemented by connecting the outputs in series. The manual actuation mode operates in parallel to independently actuate the final devices. Actuation signals are output to the loads in the form of normally de-energized, energize-to-actuate signals. The normally de-energized output state, along with the dual, 2oo2 redundancy reduces the probability of inadvertent actuation. The staff found the applicant's response to RAI-SRP-7.8-DAS-03 acceptable.

The 2oo2 DAS actuation logic was included in the approved DAS design in the certified AP1000 DCD, Revision 15. Specifically, 10 CFR 50.62 requires that ATWS mitigation equipment be designed to perform its function in a reliable manner. As described in Section 15.8.3 of the AP1000 DCD, the AP1000 is equipped with a DAS, which provides the functions required by the ATWS rule (10 CFR 50.62). The ATWS core damage frequency for the AP1000 is below the SECY-83-293, "Amendments to 10 CFR 50 Related to Anticipated Transients Without Scram (ATWS) Events," goal of 10^{-5} per reactor year. In NUREG-1793, the staff reviewed and approved the AP1000's basis for meeting the ATWS core damage frequency goal.

Reliability of digital systems can be achieved through various means including redundancy, fault detection and management, quality of design, and use of reliable components. Reliability can be defined as the likelihood that a given component or system will be properly functioning when needed, as measured over a given period of time. Reliability, in itself, does not account for any repair actions that may take place. Availability can be defined as the percentage of time that a given system will be functioning as required. In other words, availability is the probability that a system is not failed or undergoing a repair action when it needs to be used.

The AP1000 DAS design addresses reliability from a design/component approach and by fault detection and management. From a design/component reliability approach, Section 8.1 of WCAP-17184-P states that a FMEA, mean-time-between-failure analysis, and a reliability block diagram analysis will be performed on the DAS at the component level. Since the DAS detailed design is not complete at the DC stage, nor required to be complete in accordance with 10 CFR 52.47, those analyses were not part of the staff's review. However, sufficient criteria in the AP1000 DCD are available to guide the detailed design analysis, such as the use of MIL-HDBK-217F for component failures and hardware reliability analysis. From a fault detection/management approach, Section 6.1.2.2.1 of WCAP-17184-P states that the DAS will include self-diagnostic features to identify failures of the processor and supporting circuitry. The self-diagnostic features provide real-time indication to operators of a DAS failure, limiting the fault exposure time and improving DAS availability.

As part of the determination for meeting the ATWS core damage frequency goal, the AP1000 PRA assumed an availability goal for DAS, as described in Section 8.2 of WCAP-17184-P. As committed in WCAP-17184-P, the detailed reliability analysis performed on the DAS would be consistent with the availability goal. Specifically, the reliability analysis will determine an expected failure rate based on hardware failures. Both the failure rate and expected repair time will be calculated and compared to the availability goal for consistency. By utilizing self-diagnostic features, the operators are given real-time indication of a DAS failure, which allows maintenance to be performed in a timely manner. By using the self-diagnostic features, the fault exposure time is reduced on the DAS, thus improving DAS availability as it relates to latent, undetected faults.

Given the commitments in WCAP-17814-P regarding the reliability analysis and self-diagnostics, the staff finds that the DAS will operate reliably. DAS may be taken out of service for maintenance, or be subjected to a failure, but would meet the committed availability target, which is part of the overall basis for meeting the ATWS core damage frequency goal. NRC regulations such as 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," would provide verification that the availability goal is being achieved while the plant is in operation.

As part of the AP1000 design changes, the applicant proposed the addition of a new DAS high hot leg temperature reactor and turbine trip. The reason for the design change is that in the original DAS design, which is modeled in PRA, a reactor trip and turbine trip should occur for ATWS sequences with main feedwater available. Because feedwater is still available, the DAS low steam generator water level signal will not initiate the reactor or turbine trip. The DAS high hot leg temperature signal is needed to perform this function.

The staff evaluated ATWS for the AP1000 as documented in NUREG-1793. For that evaluation, the applicant analyzed a number of cases that included scenarios with and without normal feedwater operating. The most limiting case was confirmed to be the loss of normal feedwater event with turbine bypass operable, resulting in the highest RCS pressure. Addition of the hot leg temperature DAS trip would not alter that conclusion. The additional trip provides additional margin for the limiting case, and hence, is a conservative change that is acceptable for ATWS response. With respect to DAS reliability, quality, qualification, and independence from the primary protection system, the staff finds that the addition of a new function would not impact any of these design characteristics. Specifically, the function can be added into the current DAS architecture without changes to the DAS architecture described in the certified AP1000 design. Therefore, the staff finds the new DAS high hot leg temperature reactor and turbine trip meets the requirements of 10 CFR 50.62.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

7.8.3 Evaluation Findings and Conclusions

After reviewing WCAP-17184-P, Revision 2; TR-97, Revision 1; WCAP-15775, Revision 4; changes to the AP1000 DCD, Tier 1 and Tier 2; and responses to related RAIs, the staff concludes that the applicant provides sufficient information to support the changes made to the DAS in Revision 19 of the AP1000 DCD.

The staff concludes that the proposed changes to the AP1000 DCD meet the requirements of 10 CFR 50.62 and 10 CFR Part 50, Appendix A, GDC 22. The proposed changes to the AP1000 DCD did not affect the design commitments regarding an ATWS. The staff also found that the applicant provided sufficient information to support the removal of the two completed lifecycle phases and other changes for DAS. Therefore, the staff concludes the changes the applicant made to the DAS design meet regulatory requirements and criteria.

7.9 Data Communication Systems

7.9.1 System Description

The AP1000 I&C systems consist of the PMS safety system, which contains four independent divisions, four nonsafety systems, the PLS, data display and processing system (DDS), main turbine control and diagnostic system, and special monitoring system (SMS)), and two systems that perform both safety and nonsafety functions (IIS and operations and control centers system (OCS)). AP1000 DCD, Section 7.1.2.8, references WCAP-16675, Revision 5, for AP1000 data communications systems design. This TR provides an overview of the design of data communications within the PMS and communications between the PMS and nonsafety systems. Section 3 of WCAP-16675 references WCAP-16674, Revision 4, for a more detailed description of data communications in the AP1000 I&C systems design.

WCAP-16675, as supplemented by WCAP-16674, describes the use of the Common Q platform for data communications within the internal PMS safety functions (nuclear instrumentation system (NIS), QDPS, RTS, ESFAS, and the component logic system) and the safety portions of the IIS and OCS. This TR also describes the use of the Emerson Ovation® Network for data communications within the nonsafety systems, the nonsafety portions of IIS and OCS, and outputs from the safety system data (via the Advant-Ovation® Interface (AOI) gateways).

7.9.2 Communication within Safety Systems

7.9.2.1 Common Q Communications Subsystems

The NRC evaluated WCAP-16097, "Common Qualified Platform Topical Report," and issued safety evaluations approving the Common Q platform on August 11, 2000; June 22, 2001; and April 2, 2003. WCAP-16097 described the communications subsystems of the Common Q system. The SER for the Common Q platform provided an evaluation of the following three types of communications subsystems:

(1) AF100 bus communication for intrachannel communications and a separate AF100 bus for interchannel communications in the DDS

(2) HSL serial communications for interchannel communications

(3) external communications for communications between the Common Q platform and external computer systems

The Common Q platform topical report SER concluded that these three types of communications subsystems met the requirements of IEEE Std. 603-1991, as supplemented by IEEE Std. 7-4.3.2-1993. The staff's review of these communications subsystems supplements the conclusions made in the SER of the Common Q platform topical report. Specifically, the staff evaluated the application of these communications subsystems for data communications

within the AP1000 I&C systems. Sections 7.9.2.2, 7.9.2.3, and 7.9.3 document the evaluation of the application of these communications subsystems to the PMS design.

The additional detailed design information would otherwise have to be addressed through verification of implementation of the I&C DAC. Therefore, the changes to the DCD eliminate the need for I&C DAC and satisfy the finality criteria in 10 CFR 52.63(a)(1)(iv).

7.9.2.2 Intradivisional Communication via the AF100 Bus

7.9.2.2.1 Functional Description of the AF100 Bus

Section 3 of WCAP-16775 describes the use of the AF100 bus for intradivisional communications between the AC160 controllers and the safety and QDPS display systems within the same division. This section states that, within each PMS division, the internal functions and the safety portions of both IIS and the OCS are integrated using an intradivisional AF100 bus.

Specifically, the AF100 bus is used to allow the various AC160 controllers and FPDS within a division to exchange information for maintenance, test, diagnostic, communication (to the nonsafety system), display, and manual control. The majority of the dataflow is from the AC160 controllers to the FPDS (for display and for communication to the nonsafety system). Therefore, the AF100 bus is used to integrate information exchange among the AC160 controllers performing the ESFAS and reactor trip functions and the FPDS. The AF100 is not in the sensor-to-reactor trip path or sensor-to-ESFAS-actuation path. The ESFAS and reactor trip functions do not require information from each other to perform their safety functions.

The AF100 bus is a deterministic communication bus with a transmission rate of 1.5 megabit (Mbit)/second or faster. Section 4.1.2 of WCAP-16674 describes the two types of data communication that occur within the AF100 bus. The real-time data distribution communication provides a scheduled periodic broadcast of real-time data (process data transfer) pertaining to the plant processes. The general purpose communication provides a periodic exchange of data (message transfer) for other purposes, such as system operation, diagnostics, and maintenance. As described in Section 3.1 of WCAP-16675, message transfer does not influence process data transfer in any way. This is accomplished by reserving bandwidth for process data transfer and using the remaining bandwidth for message transfers. In addition, Section 3.1.1 of WCAP-16675 describes how the application program configuration tool is used to limit the maximum number of process data transfer packets that can be transferred over the AF100 bus to prevent overloading it. Process data packets are transferred with fixed packet size and cycle time to ensure deterministic communications.

Section 2.2.6.2 of WCAP-16675 states that software changes can be accomplished in the AC160 in two ways. One way is to program the AC160 over the AF100 bus. Even though this network and the only programming source (the MTP) are totally contained within a division of the PMS, this mode of programming is prevented. This is accomplished by using the AC160 Function Chart Builder tool to configure the equipment to not accept AF100 bus programming. The other way to load software into the AC160 is by a serial connection between the division's MTP and the AC160. Within a division, a separate cable is permanently routed from the maintenance and test cabinet (MTC) to each cabinet containing an AC160 processor module. This configuration allows for software loading to any processor module within a division from the MTP. The software loading cable is normally disconnected on each end. To perform a software update, the cable (coming from the cabinet containing the target processor module) in the MTC

is connected to the MTP. The opposite end of the software loading cable is connected to the target AC160 processor module and the software update is performed from the MTP.

Section 3.1 of WCAP-16675 states that an AF100 bus is totally contained within each division of the safety system. The physical extent of each AF100 bus is limited to its corresponding I&C equipment room, the MCR, and the raceways between the two. Onsite access is not provided in any other location. Offsite access to the four PMS intradivisional Common Q networks is not available. This section also states that, within the PMS, security is maintained, since the ability to remotely program the AC160 controllers and safety and QDPS display systems over the AF100 bus has been disabled in the PMS. Access to the PMS intradivisional Common Q network is only available from the MCR. Access is not available in any of the other operation and control centers.

7.9.2.2.2 Evaluation of the AF100 Bus

The staff finds the use of the Common Q AF100 bus acceptable for intradivisional communication within each PMS division. The AF100 bus only serves one division in each PMS division, with no direct connections to other divisions or nonsafety systems. Data from other divisions and nonsafety systems can only reside on the AF100 bus via the other components within the same division (e.g., integrated communications processor (ICP) through HSLs). In such cases, electrical and communications isolation is provided by the fiber-optic connection between the given component and other divisions, and the communications processor of that particular component, respectively. IEEE Std. 603-1991, Clause 5.6.1, requires independence between redundant portions of safety systems to the degree necessary to retain the capability to accomplish the safety function during and following any design-basis event requiring that safety function. In addition, IEEE Std. 603-1991, Clause 5.6.3, "Independence Between Safety Systems and Other Systems," requires independence between safety and nonsafety systems, such that credible failures in, and consequential actions by, nonsafety systems shall not prevent the safety system from accomplishing its intended safety function. Based on the communications and electrical isolation present in the Common Q AF100 bus, the staff finds the independence requirements of IEEE Std. 603-1991, Clauses 5.6.1 and 5.6.3, are met.

The staff evaluated the access controls to the AF100 bus against the requirements of IEEE Std. 603-1991, Clause 5.9, as clarified by the guidance provided in DI&C ISG #4-HICRc. IEEE Std. 603-1991, Clause 5.9, requires the design to permit the administrative control of access to safety-system equipment. DI&C ISG #4-HICRc, Section 1, Point 10, states that safety division software should be protected from alteration while the safety division is in operation. Hardwired interlocks or physical disconnection of maintenance and monitoring equipment should prevent online changes to safety-system software. The staff has evaluated the access controls described in Section 3.1 of WCAP-16675, as discussed in the above section.

The staff finds the use of the AC160 Function Chart Builder tool to configure the equipment so that it does not accept programming over the AF100 bus to be acceptable in meeting the requirements of IEEE Std. 603-1991, Clause 5.9, by addressing Section 1, Point 10, of DI&C ISG #4-HICRc regarding restrictions to the online programmability of safety equipment. In addition, the staff finds that the access control provided for programming the AC160 controller over the serial software loading cable to the equipment provides additional assurance that unauthorized software modifications to the AC160 controller and to the safety and QDPS displays are prevented. Specifically, the staff finds the physical disconnection of the software

load cable between scheduled software updates meets Section 1, Point 10, of DI&C ISG #4-HICRc.

The staff evaluated how the design of the AF100 bus addressed the system integrity requirements of IEEE Std. 603-1991, Clause 5.5, which requires in part that safety systems be designed to accomplish their safety functions under the full range of applicable conditions enumerated in the design basis. BTP 7-21, "Guidance on Digital Computer Real-Time Performance," states that risky design practices such as non-deterministic data communications, non-deterministic computation, use of interrupts, multitasking, dynamic scheduling, and event-driven design should be avoided. Based on the deterministic nature of the process data transfer on the AF100 bus, as described in Section 3.1 of WCAP-16675, and the limitations on maximum allowed process data transfer packets, the staff finds that the design of the AF100 bus adequately addresses the deterministic communications criteria provided in BTP 7-21 to meet IEEE Std. 603-1991, Clause 5.5.

7.9.2.3 Interdivisional and Intradivisional Communication via the High-Speed Link

7.9.2.3.1 Functional Description of the High-Speed Link

Section 3 of WCAP-16775 describes the use of the HSL for interdivisional and intradivisional communication within the PMS. This section states that the PMS uses HSLs, which are point-to-point, to communicate certain data within and across PMS divisions. The HSL is a serial RS 422 link using high-level datalink control protocol with a 3.1 Mbits/second transfer rate. The HSL is used for planned data exchanges of predefined data packets between two Common Q processor modules in the sensor-to-reactor trip path or sensor-to-ESFAS actuation path.

As stated in Section 6.2.1 of the SER to WCAP-16097, the PM646 function processor is divided into two sections, the process section and the communications section. The communications section in the PM646A processor module is used for HSL communications. Each processor module has one independent transmit link (output to two ports) and two independent receive links. The receivers of each HSL are independent and can receive different data independently, in accordance with the guidance of DI&C ISG #4-HICRc. As specified in Section 5.1 of the SER to WCAP-16097, these ports are used with fiber-optic cables for interchannel communications. Section 6.4 of this TR states that the integrity of data transmitted is monitored by using a CRC. The receiving processor module calculates the CRC of the received data and compares it with CRC bits received with the data. If the CRC comparison fails three consecutive times, the processor module declares the link has failed and reports the failure to the application software, which takes appropriate action. The processing section and the communication section of each PM646A processor module communicate with each other, in accordance with DI&C ISG #4-HICR, such that the communication and function processors operate asynchronously, sharing information only by means of dual-ported memory or some other shared memory resource that is dedicated exclusively to this exchange of information. This allows the two sections to share data between them while preventing either from affecting the operation of the other. The specific implementation, as described in Section 6.2.1 of the SER to WCAP-16097, is proprietary.

Based upon the specification listed in Section 4.1.1.4 of WCAP-16097, to ensure deterministic behavior of the Common Q platform, the measured load of the application programs on a single PM646 processor has to be less than 70 percent. To verify that safety systems meet the response time requirement, WCAP-16097 states that the applicant committed to perform a

throughput analysis and a response time analysis. This topical report stated that, during the testing phase of the Common Q application, the applicant will perform response time tests to validate the design's compliance with both the system response and the display response requirements. In the SER for the WCAP-16097, the staff concluded that the design features, the operation of the AC160 PLC system, and CENP's commitments to perform timing analyses and tests provide sufficient confidence that the AC160 will operate deterministically to meet the recommendations in BTP HICB-21 and is, therefore, acceptable in that regard. However, the staff issued PSAI 6.6 to ensure that timing analysis and validation tests for applications of Common Qualified platform system verify that the design satisfies the plant-specific requirements for accuracy and response time presented in the accident analysis in Chapter 15 of the safety analysis report. The resolution of PSAI 6.6 is documented in NUREG-1793 for the certified AP1000 design, which states: "The accuracy and response time of the AP1000 safety systems will be commensurate with the Chapter 15 safety analysis. The COL applicant is responsible for the setpoint analysis. The setpoint analysis shall be performed by the COL applicant, as defined in DCD Tier 1, Section 2.5.2, Item 10, and DCD Tier 2, Section 7.1.6. This is COL Action Item 7.2.7-1."

Section 3.2 of WCAP-16675 describes the functional use of the HSL for interdivisional and intradivisional communication within the PMS. Below is a summary of how the HSL is used for data communications between equipment within the PMS.

Bistable Processor Logic to Local Coincidence Logic Communication

The PMS uses the Common Q HSLs to transmit certain data for partial trip, partial actuation, and related status information calculated in the BPL controllers to the LCL controllers. In addition, these serial links are used to transmit voting information between divisions. Fiber-optic cables provide electrical isolation and the communications processor in the PM646 module provides communications isolation.

Local Coincidence Logic to Integrated Logic Processor Communication

The PMS uses Common Q HSLs to transfer ESF system-level actuation and related status information calculated in the LCL controllers to ILPs that distribute the signals to the safety components via the CIM sub-system of the PMS. These links are only used locally within a division.

Integrated Logic Processor Communication to Integrated Communication Processor Communication

The PMS uses Common Q HSLs to transfer data to support the QDPS function and data to support cross-division diagnostics between divisions. Cross-division diagnostics are completed outside the PMS, using outputs from the ICP to the PLS. For communications across divisions of the PMS, fiber-optic media converters and fiber-optic cables provide electrical isolation and the communications processor within each PM646 module provides communications isolation. Section 3.3 of WCAP-16775 states that qualified isolation devices maintain electrical isolation and communications independence between the ICP and the PLS.

Integrated Test Processor

The PMS uses Common Q HSLs to transfer data to the ITP to support testing and monitoring the PMS system. The ITP compares information from within the division via the AF100 bus to information received via HSLs from the other divisions for fault detection. Fiber-optic media converters and fiber-optic cables provide electrical isolation and the communications processor within each PM646 module provides communications isolation.

7.9.2.3.2 Evaluation of the High-Speed Link and PM646 Deterministic Performance

Clause 5.5 of IEEE Std. 603-1998 requires safety systems be designed to accomplish their safety functions under the full range of applicable conditions enumerated in the design basis. In addition, Clause 4.10 of IEEE Std. 603-1998 requires, as a part of the design basis, identification of the critical points in time or the plant conditions, after the onset of a design basis event.

To meet IEEE Std. 603-1991, Clause 5.5 and Clause 4.10, data communications systems in support of the protection system should demonstrate real-time performance in accordance with NUREG-0800 BTP 7-21. NUREG-0800 BTP 7-21 stipulates that:

1. Time delays within the data communications systems and measurement inaccuracies introduced by the data communications systems should be considered when reviewing setpoints.

2. Data rates and data bandwidths should be reviewed including impact by environmental extremes.

3. Sufficient excess capacity margins should be available to accommodate future increases.

In addition, limiting response times should be consistent with safety requirements. Digital computer timing should be consistent with the limiting response times and characteristics of the computer hardware, software, and data communications systems.

As stated above, in the SER for the Common Q topical report and NUREG-1793, the staff concluded that the HSL communications and the PM646 processor design is adequate to address the deterministic performance criteria in HICB BTP 7-21 in the Common Q topical report, and the resolution of PSAI 6.6 in NUREG-1793. The applicant included ITAAC acceptance criteria to verify the PMS response time under maximum central processing unit (CPU) loading meets Chapter 15 response time limits. Specifically, the applicant committed to modify the acceptance criteria in Item 11d) the system integration and test phase of ITAAC Table 2.5.2-8 in Tier 1, Chapter 2, Section 2.5.2 of the AP1000 DCD to state "Performance of system tests and the documentation of system test results, including a response time test will be performed under maximum CPU loading to demonstrate the PMS can fulfill its response time criteria." The staff finds this proposed update to Item 11d) of ITAAC Table 2.5.2-8 acceptable. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

IEEE Std. 603-1991, Clause 5.6.1, requires independence between redundant portions of safety systems to the degree necessary to retain the capability to accomplish the safety function during and following any design-basis event requiring that safety function. NUREG-0800 BTP 7-11,

"Guidance on Application and Qualification of Isolation Devices," states that fiber-optic cables are acceptable isolation devices. Based on the guidelines in BTP 7-11, the staff finds that optical media converters with optical fiber cabling in the HSL provide adequate electrical isolation. In addition, DI&C ISG #4-HICRc clarifies existing guidance on acceptable methods to meet the communications independence requirements of IEEE Std. 603-1991, Clause 5.6. Section 1 of this ISG specifies that communications between redundant divisions of safety systems should adhere to the points presented in that section. Table 7.9-1 documents the staff's evaluation of interdivisional communications using the HSLs against each criterion in DI&C ISG #4-HICRc.

Based on the staff's evaluation of the HSL design against the 20 criteria presented in Section 1 of DI&C ISG #4-HICRc, the staff finds that this design has addressed all criteria in Section 1 of DI&C ISG #4-HICRc, as shown in Table 7.9-1. Thus, the staff finds that the PMS design meets the independence requirements of IEEE Std. 603-1991, Clause 5.6.1.

IEEE Std. 603-1991, Clause 5.6.3, requires independence between safety and nonsafety systems, such that credible failures in, and consequential actions by, nonsafety systems shall not prevent the safety systems from accomplishing their intended safety function. Section 7.9.3 discusses communications independence between the ICP and the PLS to meet the requirements of IEEE Std. 603-1991, Clause 5.6.3.

The staff has evaluated the access controls to HSLs against requirements of IEEE Std. 603-1991, Clause 5.9. Clause 5.9 required the design to provide administrative control of access to safety-system equipment. Since the HSLs are point-to-point dedicated serial links that are only used within safety systems of the PMS, access control is maintained by the physical security controls in the MCR. The staff finds that access controls to the HSLs meet the requirements of IEEE Std. 603-1991, Clause 5.9.

7.9.2.4 CIM Communication

WCAP-17179, Revision 2, provides a description of the data communications for the CIM. Section 2.1 of this technical report states that the CIM is designed to interface a field component to the PMS and the PLS. Communication with the PMS is accomplished with the SRNC assembly. The SRNC module accepts a HSL connection. The SRNC communicates with each CIM through a safety bus known as the X bus. The X-bus is an independent, bidirectional link between the CIM and the SRNC. The PMS communication link is known as the X port. The CIMs communicate with the PLS through an Ovation® RNC. The Ovation® RNC bus is known as the Y bus.

Section 2.3.1.2.8 of WCAP-17179 states that the CIM has design features to provide for deterministic operation of the CIM. Communication between the PMS to the SRNC via the HSL is designed to be deterministic as described in Section 2.4.1.1 of this TR. Furthermore, the communication between the SRNC and the CIM via the X bus is designed for deterministic communications as described in Section 2.4.1.2 of this TR. Messages received from the PLS via the Y bus is translated to discrete digital signals prior to input into the CIM.

Evaluation of the CIM Communications

The staff evaluated how the CIM communications design addressed the system integrity requirements of IEEE Std. 603-1991, Clause 5.5, which requires in part that safety systems be designed to accomplish their safety functions under the full range of applicable conditions

enumerated in the design basis. BTP 7-21 states that risky design practices such as non-deterministic data communications, non-deterministic computation, use of interrupts, multitasking, dynamic scheduling, and event-driven design should be avoided. Based on the design commitments for deterministic operation and communications for the CIM, as stated in WCAP-17179 and summarized above, the staff finds that the applicant has adequately addressed the deterministic communications criteria provided in BTP 7-21 to meet IEEE Std. 603-1991, Clause 5.5.

Table 7.9-1. Evaluation of the PMS Interdivisional Communication via HSL

Point	Acceptability	Basis
1	The staff finds Point 1 of DI&C ISG #4 has been satisfied.	Aside from voting purposes, the components within the PMS do not receive any information from outside their division that will be used for accomplishing any safety functions.
2	The staff finds Point 2 of DI&C ISG #4-HICRc has been satisfied.	Except for voting purposes, the information received from other divisions is not used for initiating a protective function. Information received from other divisions for display and signal comparison purposes is validated through CRC checks. A communications fault in which the information is delayed, incorrect, or missing will be handled by the communications processor of each PM646 module.
3	For Point 3 of DI&C ISG #4, the applicant provided an alternative method to the guidance. The staff finds the alternative method is acceptable.	Aside from voting purposes, the information shared across divisions of the PMS does not meet DI&C ISG #4-HICRc interdivisional communication Point 3. In the PMS design, information received from other divisions for display and cross-divisional diagnostic purposes does not enhance the safety function. However, since: (1) the received information is not used for reactor trip and ESF actuation function, (2) it is communicated on a separate communication medium (AF100 bus), and (3) failure of such communication would not affect the safety functions, the staff finds the proposed alternative method acceptable.
4	The staff finds Point 4 of DI&C ISG #4 has been satisfied.	Within the PM646 module, the communications processor is separate from the function processor, and data transmission between the two processors can only be accomplished using dual port-RAM. As such, any communications errors will not propagate from the communications processor to the function processor within each PM646 module.
5	The staff finds Point 5 of DI&C ISG #4 has been satisfied.	As stated above, in the SER for the Common Q Topical Report and NUREG-1793, the staff concluded that the HSL communications and the PM646 processor design is adequate to address the deterministic performance criteria in HICB BTP 7-21 in the Common Q Topical Report, and the resolution of PSAI 6.6 in NUREG-1793. The PMS is designed to meet the Chapter 15 overall response requirements. The proposed response time testing in ITAAC Item 11d of Table 2.5.2-8 in the AP1000 DCD will verify that the as-built PMS fulfills the response time criteria under maximum CPU loading.
6	The staff finds Point 6 of DI&C ISG #4 has been satisfied.	The safety processor operates asynchronously to the communications processor within the PM646 module, and the operation does not perform any communication handshaking or accept interrupts from other divisions.

Table 7.9-1. Evaluation of the PMS Interdivisional Communication via HSL

Point	Acceptability	Basis
7	The staff finds Point 7 of DI&C ISG #4 has been satisfied.	The communication section of PM646 module only accepts predefined data sets and uses a CRC check to ensure the integrity of the data.
8	The staff finds Point 8 of DI&C ISG #4 has been satisfied.	The received information from other divisions is only used for display, diagnostic, and voting purposes. The use of predefined data sets and CRC checks ensures that the integrity of the data prevents any communications errors from affecting the safety functions.
9	The staff finds Point 9 of DI&C ISG #4 has been satisfied.	Section 2.1 of WCAP-17201 describes how the incoming messages are pre-allocated memory in static locations in the communications portion of the AC160 controller. Additional features are implemented to ensure that any software faults in the application data will be detected. Based on the storage of incoming messages in pre-allocated memory space and the use of error detection features to identify any software faults, the staff finds that Point 9 has been satisfied.
10	The staff finds Point 10 of DI&C ISG #4 has been satisfied.	As stated in Section 7.9.2.2.2 of this report, there only two methods to load software to the AC160 controllers. One is via the AF100 bus, which has been disabled during the software programming. The other way to load software into the AC160 is by a serial connection between the division's MTP and the AC160. This loading cable is normally disconnected on each end to prevent inadvertent programming during operations. As such, the staff finds Point 10 has been satisfied.
11	The staff finds Point 11 of DI&C ISG #4 has been satisfied.	The information received from other divisions is only used for display, diagnostic, and voting purposes. The design does include using the information for any other functions, including functions that allow the safety function to receive software instructions. The implementation of the design will be evaluated in the review of the specific design specifications.
12	The staff finds Point 12 of DI&C ISG #4 has been satisfied.	WCAP-17201, Section 2.2, discusses how messages will be checked to ensure validity of the message (e.g., repeated messages or messages out of sequence.) The specific features implemented to ensure the validity of the messages are proprietary. However, based on the information presented in technical report WCAP-17201, Section 2.2, the staff finds that Point 12 of ISG-04 has been satisfied.

Table 7.9-1. Evaluation of the PMS Interdivisional Communication via HSL

Point	Acceptability	Basis
13	The staff finds Point 13 of DI&C ISG #4 has been satisfied.	As described in Section 2.3 of WCAP-17201, the HSL protocol does not support error detection; only error detection is implemented to detect data communication failures. Once a message is detected as bad, it is flagged, and the application software will respond accordingly. The staff finds the use of error detection for HSL data communications acceptable to ensure received messages are correct and correctly understood. Therefore, the staff finds that point 13 has been satisfied.
14	The staff finds Point 14 of DI&C ISG #4 has been satisfied.	The HSL is a point-to-point serial link; thus, Point 14 of DI&C ISG #4 is satisfied.
15	The staff finds Point 15 of DI&C ISG #4 has been satisfied.	Although the Common Q topical report states that the communication section of the PM646 is event driven, and does not communicate a fixed set of data at regular intervals, the processing section of PM646 is cyclic. As such, as discussed in Section 2.4 of WCAP-17201, at the end of every execution cycle, the application program store data from the processing section into the dual-ported memory for the communication section to transmit. Since the communication section transmits the data whenever there is a message in dual-ported memory, this communication is thus linked to the cyclic operation of the process section. Therefore, although the communication section is event driven, the communications are really cyclic and deterministic, and thus the staff finds that Point 15 has been satisfied.
16	The staff finds Point 16 of DI&C ISG #4 has been satisfied.	The watchdog timer feature of the PM646 module ensures message and link liveliness.
17	The staff finds Point 17 of DI&C ISG #4 has been satisfied.	The AP1000 DCD and the supporting TRs, along with the Common Q topical report, and Tier 1 ITAAC have committed to completing equipment qualification.
18	The staff finds Point 18 of DI&C ISG #4 has been satisfied.	The AP1000 DCD and the supporting TRs, along with the Common Q topical report, have committed to providing the FMEA and the SHA.

Table 7.9-1. Evaluation of the PMS Interdivisional Communication via HSL

Point	Acceptability	Basis
19	The staff finds Point 19 of DI&C ISG #4 has been satisfied.	As stated above, in the SER for the Common Q Topical Report and NUREG-1793, the staff concluded that the HSL communications and the PM646 processor design is adequate to address the deterministic performance criteria in HICB BTP 7-21 in the Common Q Topical Report, and the resolution of PSAI 6.6 in NUREG-1793. The PMS is designed to meet the Chapter 15 overall response requirements. The proposed response time testing in ITAAC Item 11d of Table 2.5.2-8 in the AP1000 DCD will verify that the as-built PMS fulfills the response time criteria under maximum CPU loading.
20	The staff finds Point 20 of DI&C ISG #4 has been satisfied.	As stated above, in the SER for the Common Q Topical Report and NUREG-1793, the staff concluded that the HSL communications and the PM646 processor design is adequate to address the deterministic performance criteria in HICB BTP 7-21 in the Common Q Topical Report, and the resolution of PSAI 6.6 in NUREG-1793. The PMS is designed to meet the Chapter 15 overall response requirements. The proposed response time testing in ITAAC Item 11d of Table 2.5.2-8 in the AP1000 DCD will verify that the as-built PMS fulfills the response time criteria under maximum CPU loading.

The evaluation of the interface between the CIM and the PLS with respect to communications independence requirements, as stipulated in IEEE Std. 603-1991, Clause 5.6.3, is provided in Section 7.9.3 of this report.

7.9.2.5 Main Control Room Multiplexers

Section 7.1.2.6 of the AP1000 DCD removed the use of multiplexers in the protection and safety monitoring system to provide a signal path between the protection system equipment and the MCR. This section states that each division's safety and QDPS display will communicate with the protection system equipment via the dedicated AF100 communications network within each division. In addition, Section 3.4.1 of WCAP-16675 states that the MCR system-level actuation switches are cabled directly from the switches in the MCR to the LCL located in the bistable coincidence cabinets in each instrument room.

Since the switches in the MCR are directly connected to the LCL, the staff finds the justification for removal of the multiplexers in the MCR acceptable.

7.9.2.6 Testing of Communications Modules

The applicant removed the description of the fault tolerances, maintenance, test, and bypass from the DCD, and replaced it with references to WCAP-16675. Section 6.1 of WCAP-16675 describes the test features of communications modules within the PMS.

Sections 6.1.2 of WCAP-16675 states the AF100 bus communication modules provide communications between subsystems (e.g., BPL, LCL, ILP, MTP, and ITP). These communications include transferring data in support of system diagnostics. The AF100 bus supports two types of communications: process data and message transfer. Process data are dynamic data used to monitor and control the process, while message transfer is used for program loading and system diagnostics.

The AF100 bus communications modules are individually supervised by their own internal diagnostics and additional run-time diagnostic. In addition, the processor module performs continuous background diagnostics of the communications modules and automatically detects errors during operation. The process module contains the error messages in the error buffer for system troubleshooting.

Evaluation for Testing of Communications Modules

IEEE Std. 603-1991, Clause 5.7, requires the design to provide the capability to test and calibrate safety-system equipment, while retaining the capability of the safety systems to accomplish their safety functions. As applied to data communications systems, NUREG-0800 Section 7.9 states that data communications systems should be designed to support self-testing and surveillance testing. The design of automatic self-test features should maintain channel independence. The staff finds that the self-testing of the AF100 bus communications modules, including the internal diagnostics and additional runtime diagnostics, demonstrate conformance with the self-testing criteria in Section 7.9 of NUREG-0800 and, thus, meets the requirements of IEEE Std. 603-1991, Clause 5.7. However, the staff notes that these self-test features are not to replace the requirements for surveillance testing. In addition, since self-testing of the AF100 bus does not traverse multiple divisions or go outside the safety system, these testing features meet the independence requirements of IEEE Std. 603-1991, Clause 5.6.

7.9.3 Communication between Safety and Non-safety Systems

Section 5 of WCAP-16674 describes the data communications between the safety system and nonsafety system within the AP1000 design. This TR also describes the changes made to the certified AP1000 design regarding data communications between the safety and nonsafety systems. This TR states that the certified AP1000 design had the following data flow between the safety and nonsafety systems:

(1) data flow from PMS to PLS for control purposes
(2) data flow from PMS to DDS for information system purposes
(3) data flow from DDS to PMS for safety-system actuation purposes (via Remote Shutdown Panel when activated)
(4) data flow from PLS to PMS for component control purposes

In addition, the certified design establishes the following ITAAC in AP1000 DCD Tier 1, Section 2.5.2, regarding the implementation of these data flows:

7.a The PMS provides process signals to the PLS through isolation devices.

7.b The PMS provides process signals to the DDS through isolation devices.

7.c Data communications between safety and nonsafety systems does not inhibit the performance of the safety function.

7.d The PMS ensures that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.

In the certified design, the data flow between safety and nonsafety systems is primarily implemented using divisionalized bidirectional gateways. Section 5 of WCAP-16674 states that the data communications between the PMS and the nonsafety system have been modified in the PMS design. These modifications have the following effects:

- Reduce the dependence on the gateways.
- Make the gateways [ ].
- Create segmentation and network independence of the nuclear steam supply system (NSSS) control functions within the PLS.
- Make a clear delineation of the points of electrical, communication, and functional isolation.

In the modified design, the PMS implements data flows between safety and nonsafety equipment using divisionalized, [ ] gateways and individual analog and digital signals. Five cases of safety-system-to-nonsafety-system communication are identified within the AP1000 design. WCAP-16674 provides an analysis of the ITAAC in AP1000 DCD Tier 1, Section 2.5.2 for compliance for each of the cases, including the following requirements:

- There are isolation devices between the PMS and PLS, and between the PMS and the DDS.

- Data communications between safety and nonsafety systems do not inhibit the performance of the safety function.

- PMS ensures that the automatic safety function and the Class 1E manual controls both have priority over the non-Class 1E soft controls.

The additional detailed design information would otherwise have to be addressed through verification of implementation of the I&C DAC. Therefore, the changes to the DCD eliminate the need for I&C DAC to satisfy the finality criteria in 10 CFR 52.63(a)(1)(iv).

7.9.3.1 Description of the Five Cases of Communication between Safety and Nonsafety Systems

Below is a summary of the five cases of safety-system-to-nonsafety-system communication.

Case A and Case B

Section 5 of WCAP-16674 states that Case A and Case B communications allow the PMS to communicate with the nonsafety control system (PLS) via qualified isolation devices. Case A involves transferring safety-related input signals that are isolated in the PMS cabinets and sent to the PLS as individual hardwired analog signals. This is identical to the type of interface in existing plants. Case B allows the PMS to transfer analog and discrete digital signals calculated within the PMS to the PLS using qualified isolation devices. Section 5 of WCAP-16674 states that the qualified isolation devices used in both Case A and Case B communications provide electrical isolation between the systems as required by IEEE Standard 603-1991. They also provide functional isolation by preventing the nonsafety system from adversely affecting the safety function.

Case C

Case C communications allow various process-related signals (analog input signals, analog signals calculated within the PMS, and digital signals calculated within the PMS) to be sent to the DDS for information system (plant computer) purposes. Non-process signals, such as cabinet entry status, cabinet temperature, and direct current power supply, are also provided to the DDS for information system purposes. As described in Section 5.1.2 of WCAP-16674, the AOI Gateway in each PMS division connects that division's internal network to the nonsafety real-time data network. The sole purpose of the AOI Gateway is to provide data from the safety system to the nonsafety system for nonsafety applications. The AOI Gateway has no protection function in the PMS. The reliability of the PMS to perform its safety function is not dependent on the AOI Gateway's being functional.

The gateway has two subsystems: one is the safety subsystem that interfaces with the AF100 bus, and the other is the nonsafety subsystem that interfaces with the nonsafety Emerson Ovation® network. The AOI safety subsystem is implemented within the PMS to gain access to the desired data. This functionality is included in the PMS MTP, a Common Q FPDS. The PMS portion of the AOI function is implemented using the hardware and software dedication and qualification methodologies accepted by the NRC as part of the Common Q topical report SER process. A fiber-optic link provides communication between the safety subsystem and the

nonsafety subsystem. Communication isolation is achieved through the use of [ ] transmission of data from the optical transmitter on the safety subsystem to the optical receiver on the nonsafety subsystem.

For sequence of events (SOE) signals, such as partial trip signals, reactor trip signals, and ESFAS, each division provides the signals to the SOE system or interface via a [ ] fiber-optic link. The flow of information is strictly from the safety subsystem to the nonsafety SOE system or interface. The [ ] nature of the link is assured by the use of a single [ ] fiber. The safety end of the fiber is connected to an optical transmitter. The nonsafety end of the fiber is connected to a fiber-optic receiver. This arrangement also provides electrical isolation between the safety and nonsafety portions of the system.

Case D

Case D communications allows the nonsafety system to communicate with the safety system using discrete digital signals. These signals are used to implement nonsafety manual control of system-level safety functions (actuations, manual blocks, and resets, manual reactor trip) and a nonsafety interlock of certain PMS test functions.

Case D communications allows nonsafety manual controls of system-level safety functions that originate from dedicated switches in the RSR. Section 5.2.1 of WCAP-16674 states that in the RSR, the nonsafety manual controls of system-level safety functions (actuations, manual blocks and resets, manual reactor trip) originate from dedicated switches. The individual discrete digital signals are classified as nonsafety-related and are, therefore, isolated in the PMS cabinets before being used. At the RSR, a fiber-optic transmitter encodes the switch contact state to send over the fiber-optic cable. In the PMS, the fiber-optic receiver decodes the data and recreates the switch contact state on its discrete output signal to the AC160 rack in the safety system. Electrical isolation is provided via the fiber-optic connection. There is no metallic path to conduct an electrical fault into the PMS. Functional isolation is provided by logic within the PMS to prevent the nonsafety data flow from inhibiting the safety function. The functionality associated with these controls is disabled until operation is transferred from the MCR to the RSR. This transfer is accomplished by the divisionalized Class 1E transfer switches, which are connected directly to the LCL controllers in each division. Additionally, when the controls are enabled, their functionality is limited to that defined in the PMS functional design, because the information transferred is only in the form of discrete digital signals (i.e., there is no computer software-based communication). Specifically, the PMS design only permits the RSR manual system-level ESF actuations and the manual reactor trip inputs to initiate safety functions, not inhibit them. The manual system-level resets only remove the system-level actuation signals; they do not cause any components to change state. An additional signal is required to cause a component to change state. To reduce the chance of the spurious actuation of a function, switch contacts and communication paths are arranged in complementary pairs. Two simultaneous failures in opposite directions would be required to cause a spurious actuation.

In addition for some PMS test functions that are subject to interlocks, Case D communications also allows for transfer of discrete individual hardwired digital signals for interlocks from nonsafety equipment to the PMS. Section 5.1.2 of WCAP-16674 states that, for certain PMS test functions that are subject to interlocks from nonsafety equipment, individual hardwired digital signals from nonsafety systems are isolated in the PMS cabinets before being used. Qualified isolation devices are used. These devices provide electrical isolation between the systems, as required by IEEE Std. 603-1991.

Case E

Case E communications allows the nonsafety system to communicate with the safety system using discrete digital signals. These signals are used to implement nonsafety manual component-level controls of safety components.

As described in Section 5.2.2 of WCAP-16674, Case E communications allows manual component soft controls originating in the PLS to actuate safety components. The use of a remote I/O node, consisting of one or more Class 1E CIMs, will congregate the signals from the nonsafety manual soft controls to provide one signal digital output to a non-processor-based priority logic also contained in the CIM. The remote I/O node from the nonsafety system is physically located within each division of the safety system. The remote I/O node is electrically isolated from the nonsafety system by the fiber-optic remote I/O bus. The node is powered by the safety system, and the portions of the node not performing a safety function are qualified as an associated circuit, in accordance with IEEE Std. 384-1981. Specifically, the safety-system qualification program will demonstrate that, when it is subject to environmental, electromagnetic, and seismic stressors, it does not degrade the Class 1E circuits below an acceptable level. The environmental, electromagnetic, and seismic stressors used for these tests are the same as those used to qualify the Class 1E equipment in the same cabinet.

Within the CIM, demands from the nonsafety system are evaluated against Class 1E automatic actuation signals and Class 1E manual actuation signals from the PMS subsystem. If conflicting demands are present, the safe state of the component takes priority. The CIM uses non-processor-based priority logic hardware to implement this priority function. The CIM module also provides status updates of the safety component to the PLS. The remote I/O bus that connects the remote I/O node to the PLS uses fiber-optic cables to provide electrical isolation. As depicted in Figure 6-3 of WCAP-16674, the remote I/O bus uses bidirectional communications between the PLS and the remote I/O node. However, the communications interface of the CIM translates this data into simple discrete signals for input into the Class 1E priority logic to ensure communications independence.

PMS Interfaces to Standalone Systems

In addition to the five cases of communications between the PMS and nonsafety systems, Section 4.2.1 of WCAP-16674 states that the PMS interfaces to the standalone radiation monitoring system (RMS). RMS has two parts, one for safety functions and the other for nonsafety functions. There is no interface between the two parts. The safety portion of the RMS interfaces with the PMS using simple analog and/or discrete digital signals; this interface does not use network or datalink connections. Communication isolation does not apply to discrete hardwired signals. Electrical isolation between the RMS and the PMS is not required since the safety portion of the RMS is Class 1E. There is no interface between the nonsafety portion of the RMS and the PMS.

The CETs used by the QDPS function of the PMS are physically housed within IIS. There is no electrical interface between the CETs and the incore instrumentation electronics of the IIS. The CETs interface to the PMS using simple analog signals; these interfaces do not use network or datalink connections. Communication isolation does not apply to discrete hardwired signals. Electrical isolation between the CETs and the PMS is not required since the CETs are Class 1E.

7.9.3.2 Evaluation of Safety to Nonsafety Data Communication

IEEE Std. 603-1991, Clause 5.6.3, requires independence between safety and nonsafety systems, such that credible failures in and consequential actions by nonsafety systems shall not prevent the safety system from accomplishing its intended safety function. In addition, GDC 24 requires the protection system to be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel, which is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements for the protection system. NUREG-0800 Section 7.9 provides acceptance criteria for independence between safety and nonsafety systems to meet the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. NUREG-0800 Section 7.9 states that physical, electrical, logical, or software malfunction in one portion cannot adversely affect the safety functions of the connected system. In addition, NUREG-0800 BTP 7-11 provides acceptable methods for ensuring electrical isolation between safety and nonsafety systems. The staff evaluated each of the five cases of communication between the components within the PMS and nonsafety systems, and the PMS interface to nonsafety standalone systems, against the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. DI&C ISG #4-HICRc clarifies existing guidance on acceptable methods to meet the communications independence requirements of IEEE Std. 603-1991, Clause 5.6. Section 1 of this ISG specifies that communications between safety and nonsafety systems should adhere to the points presented in that section. The staff evaluated the safety-to-nonsafety communications scheme for each of the five cases against the criteria presented in DI&C ISG #4-HICRc. The staff's evaluation is documented below.

Case A and Case B

Based on the staff's evaluation of WCAP-16675 and WCAP-16674, the staff finds that the design of hardwired interfaces used to send analog and digital signals from the PMS to the PLS meets the electrical isolation and communications independence requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, the staff finds that since the signal transmission between the PMS and PLS is limited to analog and discrete digital signals, communications independence do not apply. Section 5 of WCAP-16674 qualified isolation devices are used in Case A and Case B communications to provide electrical isolation between the PMS and the PLS. As stated in Section 7.1.2.10 of the AP1000 DCD, isolation devices are used to maintain the electrical independence of divisions, and to prevent interaction between nonsafety-related systems and the safety-related system. Isolation devices are incorporated into selected interconnections to maintain division independence. Isolation devices serve to prevent credible faults (such as open circuits, short circuits, or applied credible voltages) in one circuit from propagating to another circuit. The staff finds these design criteria are consistent with the guidance of NUREG-0800 BTP 7-11. Since these design criteria have not changed from the certified Revision 15 of the AP1000 DCD to the current revision, the staff finds the use of qualified isolation devices acceptable to ensure adequate electrical isolation between the PMS and nonsafety systems. Therefore, the staff finds that the applicant has satisfied the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. In addition, in Tier 1, Chapter 2, Table 2.5.2-8 of the AP1000 DCD, the staff identified ITAAC for isolation devices from the PMS to the PLS and from the PMS to the DDS to ensure electrical isolation. Since these ITAAC were approved in Revision 15 of the AP1000 DCD, and no modifications were made in subsequent revisions, the staff finds that these ITAAC adequately verify electrical isolation between the PMS and the PLS and DDS to meet the electrical isolation requirements in IEEE Std. 603-1991, Clause 5.6.3, GDC 24.

Case C

The staff evaluated the description of the safety-to-nonsafety system communications in Case C against the electrical isolation requirements, and the communications and functional independence requirements, of IEEE Std. 603-1991 and GDC 24. Based on the information presented in Section 5.1.2 of WCAP-16674, the staff finds the use of one-way fiber-optical communication between the MTP and the AOI Gateway, and between the PMS and the SOE system and interface in Case C, provides adequate electrical isolation and communications independence between the PMS and the nonsafety, Ovation® network to meet the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, the staff finds that the use of fiber-optic cable for electrical isolation is in accordance with BTP 7-11. The staff finds that communications independence is achieved in the design, since the design does not include an optical receiver on the MTP for data to traverse from the nonsafety network to the PMS. Since the communication is physically [ ] from the safety system to the nonsafety system, a failure within the nonsafety system cannot propagate to the safety system. The staff finds that the safety-to-nonsafety system communications in Case C meet the communication and functional independence requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24.

Case D

The staff evaluated the description of interconnections between the RSR and the PMS in support of nonsafety manual controls of system-level safety functions in Case D communications based on the electrical isolation and functional independence requirements of IEEE Std. 603-1991, Clause 5.6, and GDC 24. Based on the information provided in Section 5.2.1 of WCAP-16674, the staff finds that the design provides adequate electrical isolation, and communications and functional independence for manual system-level ESF actuation and manual reactor trip inputs from the RSR to the PMS to meet the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, the staff makes the following findings:

- The use of fiber-optic cables provides adequate electrical isolation, as specified in NUREG-0800 BTP 7-11.

- Since the data flow from the RSR to the PMS is in the form of discrete digital signals (e.g., no communications protocol or handshaking), the guidance in DI&C ISG #4-HICR does not apply. The use of discrete digital signals for initiating system-level ESF actuation and reactor trip from the RSR using point-to-point fiber cabling provides adequate communications independence between the PMS and the RSR.

- The software within the PMS, which allows the discrete signals to only initiate safety functions, specifically, initiation of system-level ESF actuation and reactor trip, provides adequate functional isolation.

The staff evaluated the description of data communications between nonsafety equipment and the PMS in support of certain PMS test functions that require interlocks in Case D communications based on the electrical isolation and communications and functional independence requirements of IEEE Std. 603-1991, Clause 5.6, and GDC 24. The staff finds that the information presented in Section 5.2.1 of WCAP-16674 provides an adequate description of how electrical isolation, communications and functional independence are

achieved for the inputs from nonsafety equipment to the PMS to meet the requirements in IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, the staff makes the following findings:

- Electrical isolation is provided between the PMS and nonsafety equipment through the use of an isolation device. As stated above in the evaluation of Case A and Case B communications, the isolation device serve to prevent credible faults (such as open circuits, short circuits, or applied credible voltages) in one circuit from propagating to another circuit, which is consistent with the guidance of NUREG-0800 BTP 7-11.

- Since the data flow from the nonsafety equipment to the PMS is in the form of discrete digital signals (e.g., no communications protocol or handshaking), the guidance of DI&C ISG #4-HICR does not apply. The use of discrete digital signals for activating interlocks for certain PMS tests provides adequate communications independence between the PMS and the nonsafety equipment.

- The software in the PMS allows the discrete signals to only affect the ability to perform tests. The interlocks do not affect automatic or manual safety functions.

In Tier 1, Chapter 2, Section 2.5.2, Table 2.5.2-8 of the AP1000 DCD, the staff identified ITAAC for isolation devices between the PMS and the PLS and between the PMS and the DDS to ensure electrical isolation. However, the staff did not identify any ITAAC for verifying electrical isolation between the PMS and the nonsafety equipment that will be used to activate interlocks for these PMS tests. The staff requested the applicant to provide additional information to demonstrate how the qualified isolation devices provide electrical isolation between the nonsafety equipment and the PMS. Specifically, the staff requested the applicant to provide an additional ITAAC to verify electrical isolation between the PMS and the nonsafety equipment to activate these interlocks. In its response letter, dated February 8, 2010, the applicant proposed to include an additional ITAAC in Tier 1, Chapter 2, Section 2.5.2, Table 2.5.2-8 of the AP1000 DCD. The proposed ITAAC states:

> The PMS receives signals from non-safety equipment that provide interlocks for PMS test functions through isolation devices.

The proposed acceptance criterion state:

> A report exists and concludes that the isolation devices prevent credible faults from propagating into the PMS.

The staff finds the proposed ITAAC acceptable in verifying that adequate electrical isolation exists to prevent credible faults from nonsafety equipment from impacting the PMS, and thus satisfies the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. In Revision 19 to the AP1000 DCD, the applicant added Design Commitment 7e to Table 2.5.2-8, which resolves this issue.

Case E

The staff evaluated the description of data communications between the CIM and the PLS in Case E, based on the electrical isolation and communications and functional independence requirements of IEEE Std. 603-1991, Clause 5.6, and GDC 24. The staff finds that the information presented in Section 5.2.2 of WCAP-16674 provides an adequate description of how the applicant achieves electrical isolation and communications and functional

independence for nonsafety manual component-level control of safety components for Case E, to meet the requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, the staff makes the following findings:

- The use of fiber-optic cables between the remote I/O bus that connects the remote I/O node to the PLS provides adequate electrical isolation, as specified in NUREG-0800 BTP 7-11.

- Although the remote I/O bus uses bidirectional communications between the PLS and the remote I/O node, the communications interface of the CIM translates these data into simple discrete signals for input into the Class 1E priority logic. Since the data flow into the Class 1E priority logic is in the form of discrete digital signals (e.g., no communications protocol or handshaking), the guidance in DI&C ISG #4-HICR does not apply. The use of discrete digital signals for initiating nonsafety manual component-level control of safety components provides adequate communications independence between the CIM and the PLS.

The priority logic within the CIM provides functional isolation by ensuring that the PMS has priority to actuate the safety component, such that the nonsafety signal cannot prevent the PMS from actuating the component. If the nonsafety system initiates an actuation command without the PMS initiating an actuation command, then the safe state of the component takes priority.

IEEE Std. 603-1991 defines associated circuits as non-Class 1E circuits that are not physically separated or are not electrically isolated from Class 1E circuits by acceptable separation distance, safety class structures, barriers, or isolation devices. IEEE Std. 384-1992, Clause 5.5, includes the classification and qualification of associated circuits. IEEE Std. 384-1992, Clause 5.5.3, states that associated circuits, including their isolation devices or the connected loads without the isolation devices, shall be subject to the qualification requirements placed on Class 1E circuits to ensure that the Class 1E circuits are not degraded below an acceptable level. Associated circuits need not be qualified for performance of function, since the function is non-Class 1E. The staff finds that Section 5.2.2 of WCAP-16674 adequately demonstrates how the RNC is qualified as an associated circuit to meet the requirements of IEEE Std. 603-1991, Clause 5.6, and IEEE Std. 384-1992. Specifically, the staff finds that the commitment, as part of the overall safety-system qualification program, to demonstrate that, when the RNC is subject to environmental, electromagnetic, and seismic stressors, it does not degrade the Class 1E circuits below an acceptable level, meets the associated circuit qualification requirements in IEEE Std. 384-1992, Clause 5.5.3.

PMS Interfaces to Standalone Systems

Section 4.2.1 of WCAP-16675 provides a description of the PMS interfaces to standalone safety systems. This section states that the PMS interfaces to the standalone RMS. However, there is no interface between the safety portion of the RMS and the nonsafety portion.

In addition, the CETs used by the QDPS function of the PMS are physically housed within the IIS. There is no electrical interface between the CETs and the incore instrumentation electronics of the IIS. The CETs interface to the PMS using simple analog signals; these interfaces do not use network or datalink connections.

Based on the information provided in Section 4.2.1 of WCAP-16675 regarding PMS interfaces with standalone systems, such as the RMS, the staff finds the design adequately demonstrates

compliance with the communications and functional independence requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24. Specifically, for the RMS, since there is no interface between the safety portion of the RMS and the nonsafety portion, communications independence does not apply. In addition, functional independence is achieved through the isolation of safety functions to only the safety portion of the RMS, and does not require interaction with the nonsafety portion to perform the intended safety functions. For the CETS used by the QDPS function of the PMS, the staff finds that since the CETS interface to the PMS using simple analog signals and these signals are Class 1E, communication independence does not apply. Since the CETS are Class 1E, the staff finds that electrical isolation between the CETS and the PMS is not required.

7.9.4 Nonsafety Communications

7.9.4.1 Description of the Nonsafety Communication Network

Nonsafety communications consist primarily of the nonsafety communication network and the nonsafety data link interface. The nonsafety communication network is implemented using the Ovation® network. This network uses unaltered Ethernet protocols, high-speed Ethernet switches, and full duplex cabling (fiber or copper shield twisted pair).

Section 3 of WCAP-16674 provides a detailed description of the AP1000 nonsafety communications system. The nonsafety communications network provides real-time data distribution and general purpose communications. Real-time data distribution is defined as the scheduled periodic broadcast of real-time data pertaining to the plant processes. The term "general purpose communications" is defined as the aperiodic exchange of data for other purposes, such as system operation, diagnostics, and maintenance.

The Ovation® network supports network standard communications protocols, such as Transmission Control Protocol/Internet Protocol and User Datagram Protocol/Internet Protocol for general purpose communications. Within the Ovation® system, general purpose communications based on standard protocols are used for aperiodic data, including file-type data transferred from the historian and plant databases to be presented at the HSI, plant informational data messages, alarm messages, and SOE messages to the plant historian for long-term historical storage. This communication occurs on the same physical media as the real-time periodic data, but it is implemented in such a way as to preserve the design philosophy of guaranteeing the real-time periodic data transmission without loss, degradation, or delay, even during plant upsets.

With respect to periodic data, the network is designed to support up to 200,000 point values per second, using a nominal percentage of the overall network bandwidth. The network load associated with periodic data origination is constant; it does not change during plant upset conditions. The Ovation® vendor has tested the network at the limit of 200,000 point values per second. A design goal is to limit the number of point values per second, to the extent possible. This will provide additional spare capacity and will result in a lower base load on the network. As the design is finalized, a firm number of point values per second will be determined. This will be used to calculate the base network load and, therefore, the network bandwidth available for aperiodic data communications. With respect to aperiodic data, the network load is variable but managed. The aperiodic data levels can be managed through careful system configuration. Alarm message data will be minimized, to the extent possible, by limiting the number of points subject to alarm checking and by carefully selecting alarm limits to minimize nuisance alarms. Network impacts associated with station staff in the main control area are somewhat limited by

the number of operators and operator stations and by the number of engineers and engineering stations. In general, the network load from aperiodic data traffic is expected to be very small in relation to the overall bandwidth of the system. Analytical justification of network capacities will be reviewed for correctness. Based on the current evaluation of expected network traffic, the single network design will meet or exceed all system capacity and network loading requirements.

Storm control is configured on the Ovation® network to ensure that highway availability requirements are satisfied, given the possibility that a software or hardware malfunction, or a malicious network attack, would introduce a packet storm on the control system highway. Storm control is implemented with configuration settings provided by the switch operating system. In general, each port subject to storm control is configured with traffic ingress block and restoration settings. These values are typically a percentage of the total available bandwidth that can be used by the broadcast or multicast traffic. When traffic entering a port exceeds the predefined block value, packet forwarding on the port is blocked. Packet forwarding resumes when the traffic falls below the predefined "restore forwarding" setting. Storm control is put in place to protect the network from data storms produced as a result of atypical conditions, including hardware malfunctions, and errors introduced by humans. The thresholds are set on a per-port basis, so that native Ovation® traffic (e.g., periodic process point data, aperiodic alarm message traffic) will not activate the storm control function. In addition to the system storm control configuration installed on the network switches, the Ovation® controller has been hardened against excessive network traffic through the use of a software modification that prioritizes critical control functionality over network communications.

7.9.4.2 Evaluation of the Non-safety Communication Network

The staff evaluated the adequacy of the nonsafety Ovation® network to perform the required control functions specified in the AP1000 DCD and the supporting TRs, as well as WCAP-16675 and WCAP-16674, including how the applicant met the requirements of 10 CFR 52.47(a)(9). Regulations in 10 CFR 52.47(a)(9) require applications for light-water-cooled NPPs to evaluate the standard plant design against the NUREG-0800 revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in the NUREG-0800 acceptance criteria. NUREG-0800 Section 7.9 provides performance criteria for data communication systems; specifically, for system capacity, data rates, and bandwidth requirements. The staff finds that the description provided for system capacity, data rates, and bandwidth requirements, and the analysis on expected network traffic, presented in Section 3 of WCAP-16674, adequately address the performance criteria for data communications systems specified in NUREG-0800 Section 7.9. Specifically, the staff finds that the evaluation of expected network traffic demonstrates that the network design is bounded by the Ovation® system capacity and network loading requirements.

Specifically, the staff evaluated the nonsafety Ovation® network design features to determine how the applicant has addressed operating experience with data storm, as described in NRC Information Notice 2007-15, "Effects of Ethernet-Based, Non-Safety Related Controls on the Safe and Continued Operation of Nuclear Power Stations," dated April 17, 2007. The staff finds that the storm control features within the Ovation® network design adequately demonstrate how data storms are precluded in the Ovation® network. Specifically, the staff finds that the use of storm control configuration settings provided by the switch operating system, and the software feature that prioritizes critical control functionality over network communications, ensure that the

Ovation® controller can continue to control critical plant operations during a network storm or a complete loss-of-network event.

7.9.4.3 Description of the Non-safety Data Link Interfaces

Section 3.2 of WCAP-16674 describes the nonsafety data link interfaces in the AP1000 I&C design. Each system is summarized below.

Standalone Systems

The Ovation® system supports standard and custom data links, both at the controller and workstation level. Controller-level interfaces include standard interfaces to Allen-Bradley programmable logic controllers and GE Mark V/VI, Toshiba, and MHI turbine control systems, as well as a standard MODBUS interface and OSI PI historian interface. At the controller level, the data link interface can be accomplished via a standard I/O module (the R-line Link Controller), or via fast Ethernet communications interfaces at the controller processor level.

Remote I/O

The Ovation® system supports the use of remote I/O, so that I/O modules can be clustered close to field devices, minimizing field cabling costs and also accommodating harsher environments. Remote I/O is in contrast to local I/O, which is housed in the same cabinet as the controller or next to it in an extended cabinet. For local I/O, all I/O modules reside in up to four cabinets, which are placed side by side. All field wiring leads to these cabinets.

Non-safety Smart I/O Field Buses

The Ovation® system supports HART I/O, FOUNDATION™ Fieldbus, Prefabs DP, and DeviceNet™ smart I/O interfaces. The Ovation® fieldbus solution is modular, and a single controller can simultaneously interface to fieldbus devices, HART I/O modules, conventional I/O modules, and third-party I/O.

HART I/O

The Ovation® controller supports native HART I/O modules. HART is technology that provides a digital information signal superimposed on a 4-20 milliampere traditional sensor loop. The digitized signal provides up to four HART multivariables, which provide additional information from HART-enabled devices, eliminating additional cabling required to provide the same information using traditional sensors and control output devices.

The Ovation® HART input module has eight inputs, with each input having an individual HART modem (supporting up to four HART multivariables), and individual channel-to-channel isolation. The Ovation® HART output module has four channels, also with individual HART modems per channel, and individual channel-to-channel isolation.

Foundation Fieldbus

FOUNDATION™ Fieldbus H1 is typically used for analog devices, such as sensors and modulating control valves. A large assortment of "smart" devices is available with the interface. The Ovation® FOUNDATION™ Fieldbus solution is modular and scalable. The interface between the FOUNDATION™ Fieldbus instrumentation and the Ovation® controller is via

dedicated, redundant Fieldbus Ethernet switches, and Ovation® FOUNDATION™ Fieldbus Gateways. There are up to 16 FOUNDATION™ Fieldbus gateways per controller, with up to four H1 segments per gateway and up to 16 devices per segment.

Profibus DP

Profibus DP is typically used for digital on/off devices. In addition to being supported by the appropriate devices, it is suitable for long distances while remaining less sensitive to power, grounding, polarity, and resistance concerns.

The Ovation® Profibus Interface uses a standard Ethernet switch, attached to the Ovation® controller via the controller's standard MODBUS/TCP third-party I/O capability.

DeviceNet

DeviceNet™ is an interface for discrete actuators and sensors. The Ovation® DeviceNet™ interface has the same fundamental design as the Profibus DP interface, using a standard Ethernet switch, attached to the Ovation® controller via the controller standard MODBUS/TCP third-party I/O capability.

Asset Management

Another important component of the intelligent field interface solution is the Asset Management Solutions (AMS) suite of software. AMS software and the associated SNAP-ON applications is a suite of software solutions for streamlining all maintenance activities related to instrumentation and valves in a process plant. This package can be integrated into the Ovation® workstation and Ovation® controller to give the user direct access to all intelligent devices connected to the Ovation® I/O. With AMS integrated into Ovation®, digitized HART or FOUNDATION™, Fieldbus parameters such as valve position can be mapped to Ovation® process points that can be used anywhere they are required in the Ovation® distributed control system. AMS provides direct visibility from the Ovation® workstation to each "smart" device in the plant that is connected to Ovation®.

7.9.4.4 Evaluation of the Nonsafety Data Link Interfaces

The staff evaluated the nonsafety data link interfaces within the AP1000 I&C design. IEEE Std. 603 1991, Clause 5.6.3, requires independence between safety and nonsafety systems so that credible failures in, and consequential actions by, nonsafety systems shall not prevent the safety system from accomplishing its intended safety function. In addition, GDC 24 requires the protection system to be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel that is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. These nonsafety data link interfaces do not communicate with any of the safety systems beyond the five cases of safety-to-nonsafety system data communications specified in WCAP-16674. As such, the staff finds that the electrical isolation, communications and functional independence requirements of IEEE Std. 603-1991, Clause 5.6.3, and GDC 24 do not apply.

7.9.5 Secure Development and Operational Environment

10 CFR Part 50, Appendix A, GDC 21 requires, in part, that protection systems (or safety systems) must be designed for high functional reliability commensurate with the safety functions to be performed. Criterion III of Appendix B to 10 CFR Part 50 requires, in part, that quality standards must be specified and design control measures must be provided for verifying or checking the adequacy of design.

10 CFR 50.55a(h) requires that safety systems for NPPs must meet the requirements stated in IEEE Std. 603-1991. Clause 5.6.3 of IEEE Std. 603-1991 requires safety systems to be designed, such that credible failures in and consequential actions by other systems will not prevent safety systems from performing their intended safety functions. In addition, Clause 5.9 of IEEE Std. 603-1991 requires the design to permit the administrative control of access to safety system equipment. These administrative controls shall be supported by provisions within the safety systems, by provision in the generating station design, or by a combination thereof.

RG 1.152, "Criteria for Digital Computers in Safety Systems of Nuclear Power Plants," Revision 2, provides a method that the NRC finds acceptable for complying with the Commission's regulations (i.e., 10 CFR Part 50, Appendix A, GDC 21, Criterion III of Appendix B to 10 CFR Part 50, and IEEE Std. 603-1991, Clauses 5.6.3 and 5.9) for promoting high functional reliability, design quality, and security for use of digital computers in safety systems of NPPs. Security in this context refers to the establishment of an SDOE for digital safety systems by: (i) measures and controls taken to establish a secure environment for development of the digital safety system against undocumented, unneeded and unwanted modifications; and (ii) protective actions taken against a predictable set of undesirable acts (e.g., inadvertent operator actions or the undesirable behavior of connected systems) that could challenge the integrity, reliability, or functionality of a digital safety system during operations. RG 1.152, Revision 2, utilizes the waterfall life cycle phases to provide a framework for establishing digital safety system security guidance, as well as criteria for acceptability, in the development of high quality safety systems.

As proposed in the response to Open Item OI-SRP7.1-ICE-01, Section 7.1.2.14.1 of the AP1000 DCD will be revised to reference, APP-GW-J0R-012, hereinafter referred to as the "PMS Computer Security Plan," to demonstrate how computer security is incorporated into the design and development of AP1000 safety systems. This plan provides a description of the planning phase for the AP1000 PMS. This plan summarizes the quality standards and design control measures implemented to provide computer security and ensure that the PMS and CIM are designed for high functional reliability commensurate with the safety functions to be performed throughout the development phases of digital safety system lifecycle. These commitments include the design and development of security features and development controls for the PMS and CIM. Although the PMS Computer Security Plan does not officially commit to conformance to RG 1.152, Revision 2, this TR addresses the system security aspects of the Common Q platform, PMS application, and CIM from the concepts phase through the test phase to protect against non-malicious events that is consistent with the criteria provided by RG 1.152, Revision 2.

The safety evaluation for AP1000 I&C secure development and operational environment, provides a separate assessment of the PMS Computer Security Plan and includes official use only information. This separate safety evaluation forms the basis for the following conclusions.

- The identified vulnerabilities in the PMS design for the conceptual phase of the development life cycle, and the security capabilities that mitigate these vulnerabilities adequately address Regulatory Position C.2.1 in RG 1.152, Revision 2.

- The performance of additional V&V activities for the Common Q platform satisfies the criteria for identifying and mitigating vulnerabilities as specified in Regulatory Position C.2.1 of RG 1.152, Revision 2.

- Although the vulnerabilities of the PMS and CIM development process are general, these vulnerabilities were only based on the conceptual phase assessment, and these vulnerabilities encompass more detailed vulnerabilities that may be identified in later portions of the development process. Thus, the identified vulnerabilities are acceptable in specifying particular portions of the development process that are susceptible to unintended or inadvertent modification to the PMS or CIM while under development or to the development tools. As such, the applicant has adequately addressed the criteria in Regulatory Position C.2.1 of RG 1.152, Revision 2, for the PMS application and the CIM. In addition, the quality assurance program for both the development of the PMS and the CIM, and the V&V process are adequate to mitigate the identified vulnerabilities by identifying and preventing inadvertent changes to the PMS and CIM design during development.

- By precluding capabilities for remote access to the PMS during operations in the design and by ensuring one way data flow from the PMS to nonsafety systems (except for use of discrete digital or analog signal), the applicant has satisfied Regulatory Position C.2.1 in RG 1.152, Revision 2. In addition, the commitment to ensure that the isolated development infrastructure (IDI) is created to preclude remote access is sufficient to satisfy Regulatory Position C.2.1 in RG 1.152, Revision 2.

- Based on the PMS access control functional requirements, the safety to nonsafety interfaces requirements, and the commitment to ensure that proper human factors are considered during the development of the PMS design, these requirements adequately provide sufficient measures to protect the PMS from inadvertent operator actions or unpredictable behavior of connected systems during operations to address the criterion in Regulatory Position C.2.2.1 of RG 1.152, Revision 2, to define the security functional performance requirements.

- The Common Q Platform has adequately addressed the criteria in Regulatory Position C.2.2.1 of RG 1.152, Revision 2. Specifically, the Common Q platform software V&V, as described in Section 5.5.3 of the Common Q Software Program Manual, was reviewed and accepted by the NRC in the SER for the Common Q platform SPM. This includes the approval of the V&V process for use of pre-developed software within the Common Q platform. The V&V activities that were performed on the Common Q platform are adequate to ensure the integrity, reliability, and functionality of this platform for use in the AP1000 PMS application.

- By incorporating the security requirements into the overall system requirements, the PMS V&V process in accordance with the Common Q SPM is adequate to ensure the correctness, completeness, accuracy, testability, and consistency of the system security requirements. Thus, the applicant has met the criteria in Regulatory Position C.2.2.1 of RG 1.152, Revision 2.

- Based on the described security measures provided in the IDI, the commitment to assess and mitigate vulnerabilities in the IDI, and the quality assurance and V&V process described in the PMS Computer Security Plan, the applicant has adequately ensured that undocumented code or functions are precluded in the design to meet Regulatory Position C.2.2.2 of RG 1.152, Revision 2.

- By incorporating the security features as part of the overall system design, the design process described in Section 2.3 of the PMS Computer Security Plan is adequate to ensure that the system security requirements is accurately translated into specific design configuration items to meet Regulatory Position C.2.3.1 of RG 1.152, Revision 2. In addition, the additional security assessment completed during the design phase is adequate to ensure that any vulnerability that has not been identified during earlier phases of the development life cycle is captured and that the security features chosen in the conceptual phase are adequate.

- Based on the quality assurance and V&V process described in the PMS Computer Security Plan, the staff finds that the applicant has adequately ensured that undocumented code or functions are precluded in the PMS and CIM design to meet Regulatory Positions C.2.3.2 of RG 1.152, Revision 2. This is based on control of the design document revision process, storage of design process in an accessed controlled manner, and requirements traceability to ensure that all design features are traceable to requirements specifications.

- By incorporating the security features as part of the overall system implementation, the implementation process described in Section 2.4 of the PMS Computer Security Plan is adequate to ensure that the system design is accurately transformed into code, database structures, and related machine executable representations to meet Regulatory Position C.2.3.1 of RG 1.152, Revision 2. In addition, the additional security assessment completed during the implementation phase is adequate to ensure that the security controls chosen are adequate.

- Based on the commitment to secure the IDI, to perform testing and scanning to identify undocumented code and functions, and to follow the quality assurance and V&V process described in Section 2.1.2.2 of APP-GW-J0R-012, Revision 1, the applicant has adequately ensured that undocumented code or functions are precluded in the PMS and CIM implementation to meet Regulatory Positions C.2.4.2 of RG 1.152, Revision 2.

- Based on the commitment to perform integration, system, and acceptance tests where practical and necessary on the PMS security features, including testing of the system configuration, and the performance of additional vulnerability assessments to ensure that no new vulnerabilities are identified in the PMS, the applicant has adequately addressed Regulatory Position 2.5 of RG 1.152, Revision 2.

- Based on the commitment to secure the testing environment and to test the hardware architecture, external communication devices, and configurations for unauthorized pathways that affect system integrity, the applicant has adequately addressed Regulatory Position C.2.5.2 of RG 1.152, Revision 2.

Based on the staff's conclusions on the PMS Computer Security Plan, as discussed above, the staff finds that the applicant has sufficiently addressed the criteria in Regulatory Positions C.2.1 through C.2.5 of RG 1.152, Revision 2, to meet the requirements of 10 CFR Part 50, Appendix A, GDC 21; Criterion III of Appendix B to 10 CFR Part 50; and IEEE Std. 603-1991, Clauses 5.6.3 and 5.9; as it relates to security and reliability of the PMS application and CIM. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

7.9.6 Evaluation, Findings, and Conclusions

The staff reviewed the revisions to Section 7.1 and the associated TRs of the AP1000 DCD against the regulatory requirements of a data communications system as stipulated in the guidance of NUREG-0800 Section 7.9. Below is a summary of the staff's findings.

Regulations in 10 CFR 50.55a(h) require compliance with IEEE Std. 603-1991 and the correction sheet, dated January 30, 1995. The minimum requirements that are applicable to all data communications systems are in IEEE Std. 603-1991, Clause 5.6.3. Other criteria include those in Clauses 5.4, 5.6.1, 5.7, and 5.9. The staff evaluated the data communications systems in the amendments to the AP1000 I&C systems design for conformance to the requirements of IEEE Std. 603-1991 and has the following findings:

- IEEE Std. 603-1991, Clause 5.4: This requirement has been fully satisfied, as documented in Section 7.9.3. In Table 2.5.2-8 of the AP1000 DCD, the staff identified ITAAC for the seismic, environmental, and Class 1E qualification of PMS equipment, including equipment used for data communications in the PMS.

- IEEE Std. 603-1991, Clause 5.6.1: This requirement has been fully satisfied, as documented in Section 7.9.2.3. The staff finds that the 20 criteria presented in Section 1 of DI&C ISG #4-HICRc have been satisfied in the design for interdivisional communication between divisions of the PMS.

- IEEE Std. 603-1991, Clause 5.6.3: This requirement has been fully satisfied, as documented in Section 7.9.3. The staff finds that the information presented in the AP1000 DCD and the supporting TRs have sufficiently demonstrated how independence is achieved for each of the five cases of safety and nonsafety communications.

- IEEE Std. 603-1991, Clause 5.7: This requirement has been satisfied, as documented in Section 7.9.2.5.

- IEEE Std. 603-1991, Clause 5.9: This requirement has been fully satisfied, as documented in Section 7.9.2. The AP1000 DCD and the supporting TRs have addressed how access controls are incorporated into the design of the PMS.

Regulations in 10 CFR 52.47(a)(9) require applications for light-water-cooled NPPs to evaluate the standard plant design against the NUREG-0800 revision in effect 6 months before the docket date of the application. The evaluation required by this section shall include an identification and description of all differences in design features, analytical techniques, and procedural measures proposed for the design and those corresponding features, techniques, and measures given in the NUREG-0800 acceptance criteria. NUREG-0800 Section 7.9 provides the design considerations for data communications systems, including criteria for

performance and reliability considerations. The staff evaluated the data communications systems in the AP1000 DCD against the guidance provided in NUREG-0800 Section 7.9, which states that digital computer timing should be consistent with the limiting response times and characteristics of the computer hardware, software, and data communications systems. The staff found the applicant's commitment to modify the acceptance criteria in Item 11d) of ITAAC Table 2.5.2-8 in the AP1000 DCD to state "Performance of system tests and the documentation of system test results, including a response time test will be performed under maximum CPU loading to demonstrate the PMS can fulfill its response time criteria" is acceptable to address the criteria in NUREG-0800 Section 7.9 and, therefore, meets the requirements of 10 CFR 52.47(a)(9).

GDC 21 requires the protection system to be designed for high functional reliability and inservice testability, commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to ensure that: (1) no single failure results in loss of the protection function; and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy, unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic tests of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy. Based on the staff's conclusions of the PMS Computer Security Plan, as discussed in Section 7.9.5 of this report, the staff finds that the applicant sufficiently addressed the criteria in Regulatory Positions C.2.1 through C.2.5 of RG 1.152, Revision 2 to meet the requirements of 10 CFR Part 50, Appendix A, GDC 21; Criterion III of Appendix B to 10 CFR Part 50; and IEEE Std. 603-1991, Clauses 5.6.3 and 5.9, as it relates to security and reliability of the PMS application and CIM. As such, the staff concludes that the AP1000 design meets the requirements of 10 CFR Part 50, Appendix A, GDC 21; Criterion III of Appendix B to 10 CFR Part 50; and IEEE Std. 603-1991, Clauses 5.6.3 and 5.9; as it relates to security and reliability of the PMS application and CIM.

GDC 24 requires the protection system to be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel that is common to the control and protection systems, leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Based on the review of the interfaces between the PMS and the PLS, the staff concludes that this requirement has been fully satisfied, as documented in Section 7.9.3. The staff finds that the information presented in the AP1000 DCD and the supporting TRs have sufficiently demonstrated how independence is achieved for each of the five cases of safety and nonsafety communications.

The additional detailed design information for the I&C architecture and communications results in increased standardization of this aspect of the design. Therefore, the change meets the finality criterion for changes in 10 CFR 52.63(a)(1)(vii).

APPENDIX 7.A: EVALUATION OF APP-GW-GLR-137, REVISION 0, "BASES OF DIGITAL OVERPOWER AND OVERTEMPERATURE DELTA-T (OPΔT/OTΔT) REACTOR TRIPS"

7.A.1 Introduction

This safety evaluation addresses changes made from Revision 15 to Revision 19 of the AP1000 DCD regarding a change in methodology for the thermal overtemperature delta-temperature (OTΔT) and thermal overpower delta-temperature (OPΔT) reactor trip design bases. Revision 15 of the DCD references WCAP-8745-P-A, "Design Bases for the Thermal Overpower ΔT and Thermal Overtemperature ΔT Trip Functions." This is the previously approved topical report for the analog calculation of the reactor trip functions. Revision 19 also references WCAP-8745-P-A; however, Revision 19 includes a change from an analog-based OTΔT and OPΔT design to a digital-based design with a different calculational methodology for the trip function. The basis of the setpoint calculations is unchanged from that presented in WCAP-8745-P-A, but the inputs to both margin-to-trip functions have changed. The digital-based core power indication described in APP-GW-GLR-137, Revision 1, proposes the use of density at the reactor core inlet and the enthalpy difference between the exit and inlet of the core, referred to as the "ΔT power signal," to provide a more accurate measurement of core power. The new TR also claims the setpoint functions have been simplified.

7.A.2 Evaluation

7.A.2.1 Background

RAI-SRP16-CTSB-42 requested that the applicant either submit a previously approved reference supporting the changes to the OTΔT and OPΔT trip functions or submit a reference that supports the changes. The RAI also requested that the applicant comply with generic letter (GL) 88-16, "Guidance for Technical Specification Changes for Cycle-Specific Parameter Limits," to include the appropriate and approved methodology regarding the revised trip functions in Technical Specification (TS) Section 3.3.1-1 and bases, which are addressed in Chapter 16 of this report.

The response to RAI-SRP16-CTSB-42 led to the generation of Open Item OI-SRP16-CTSB-42, since the response did not fully address the staff's request. SER Chapter 16, which is based on changes to the DCD between Revision 15 and Revision 17, references the open item.

The initial response to Open Item OI-SRP16-CTSB-42 was submitted to address staff concerns. A reference to APP-GW-GLR-137, Revision 1, was given to provide the information requested by the staff. Changes to the DCD for Revision 17 were also provided to maintain consistency between the submitted TR and the following DCD sections:

- Section 7.2.1.1.3
- Section 7.2.4
- TS Table 3.3.1, Notes 1 and 2

The staff submitted RAIs based on the review of APP-GW-GLR-137. After reviewing the responses, the staff found the information given in APP-GW-GLR-137 to be acceptable in support of the digital-based OPΔT and OTΔT reactor trip function methodology.

7.A.2.2 Proposed Change

The OPΔT and OTΔT trips are used to protect the specified acceptable fuel design limits (SAFDLs) so as to maintain the fuel in a geometry amenable to cooling. The design basis of the OTΔT trip is to prevent a departure from nucleate boiling (DNB) on all fuel surfaces, while the design basis of the OPΔT trip is to prevent excessive fuel centerline temperatures for all fuel rods.

In the analog technology of the OPΔT trip setpoint, as described in WCAP-8745-P-A, the setpoint is calculated as a function of the average coolant temperature (T_{AVG}) and a core power reduction term related to adverse axial offset. The OTΔT setpoint has the same inputs as the OPΔT trip setpoint with the addition of pressurizer pressure. The setpoints use T_{AVG}, axial offset, and pressurizer pressure (only for OTΔT) as inputs to a dynamically compensated function to determine the percent of rated thermal power (RTP) at which the reactor should trip. The analog signals are converted to ΔT signals by adjusting gains. The basis for the determination of both ΔT setpoints is derived from thermal design limits as explained in WCAP-8745-P-A.

The change from the analog technology to the digital technology is in both the reactor trip function and the RTP measurement. The analog method uses ΔT as a measure of core power, while the digital method uses actual core power. In the digital method, core power is determined by calculating an enthalpy difference between the core inlet and outlet. The inputs from the respective protection system divisions used to calculate the enthalpy terms are T_H for the outlet, T_C for the inlet, and pressurizer pressure, which is used in both terms. Core average temperature is eliminated as the major functional variable in the digital-based function.

In both the analog and digital technology, the OPΔT trip setpoint uses only a preset bias based on a percentage of RTP, which is based on pre-determined thermal limits, as discussed in WCAP-8745-P-A. This term is compared to core thermal power and then dynamically compensated to obtain a margin to trip signal.

In the proposed change to digital technology, the OTΔT trip setpoint directly translates DNB thermal design limits, which give core inlet temperature as a function of RTP for various pressurizer pressures, into inputs for the setpoint calculation. Core inlet temperature and pressurizer pressure information are linearly interpolated from a table that provides the corresponding RTP to serve as the appropriate setpoint. This setpoint is compared to the core thermal power calculation and then dynamically compensated to obtain a margin to trip signal.

7.A.2.3 Regulatory Basis

10 CFR Part 50, Appendix A, GDC 10, "Reactor Design," states: "The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences." GDC 10, therefore, applies directly to the design of the OTΔT and OPΔT reactor trips since they are part of the reactor protection system.

Furthermore, GL 88-16 was issued to allow licensees to update all applicable cycle-specific limits without formal review by the NRC. These cycle-specific limits are now located in the core operating limits report (COLR) and are referenced throughout TS. The methodologies by which

the cycle-specific limits are updated undergo a formal review by the NRC and are referenced in Section 3.3.1 and bases of the TS.

Since the applicant has revised the method by which the OPΔT and OTΔT reactor trip functions are calculated in the digital methodology, as discussed in APP-GW-GLR-137, the new methodology must be reviewed and approved by the staff. Furthermore, once the review is completed and approved, TS 3.3.1 and bases should be updated to reflect the methodologies that serve as the basis for determining the limits given in the COLR.

7.A.2.4 Evaluation

As previously discussed, all methodologies used to update limits given in the COLR must be reviewed and approved by the staff. The change in calculational methodology of the OPΔT and OTΔT reactor trip setpoints is given in APP-GW-GLR-137. The staff reviewed the document and submitted RAIs to better understand how the trip function calculations have changed and to ensure that SAFDLs will not be exceeded.

A review of the RAI responses in support of Open Item OI-SRP16-CTSB-42 closure is provided in the following discussion.

- Question 1 asks how the single failure criterion of 10 CFR 50.55a(h) is met when an individual resistance thermowell detector (RTD) in one of four divisions votes for a trip due to an approach to a saturation condition. In the response, it is stated that the single failure criteria is not impacted by an RTD in a saturated condition since the 2oo4 voting logic by division is unaffected. When a single RTD approaches saturation, it is removed from the average hot leg temperature calculation for that division. Further, it was stated that when two RTDs approach saturation, a trip vote occurs in the affected division. Based on the applicant's response, it was determined that 10 CFR 50.55a(h) has not been violated and Question 1 is resolved.

  Question 2 discusses the accuracy of the ΔT power signal and asks how the bias applied to the Thot-local signal leads to approximation of the mixed mean hot leg temperature. The concern is that the correction factor applied to the individual hot leg RTDs, which are used to calculate the average hot leg temperature, might lead to an inaccurate calculation of the ΔT power signal. The response states that the ΔT power signal is frequently calibrated, as required by TS;, consequently there is reasonable assurance that the streaming bias applied to the Thot-local signals will not affect the calculation of the ΔT power signal used in the margin to trip calculation. Based on the applicant's indication that the ΔT power signal is continuously monitored and validated, Question 2 is resolved and the applicant indicated that the TR will be updated for clarification. These changes were incorporated in a subsequent revision to the TR.

- Question 3 is related to the discussion of the redundant sensor algorithm and asks if the discussion is included in Chapter 7 of the DCD. The applicant stated that APP-GW-GLR-137, which includes the discussion, is referenced in Chapter 7 of the DCD. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- Question 4 is concerned with how the weighted averaging of the $T_{hot\text{-}local}$ signals is performed for T_H determination. Specifically, the concern is that automatic adjustment of weighting factors might violate 10 CFR Part 50, Appendix B, Criterion III, "Design

Control." The response states that weighting factors are not changed. If one of the $T_{hot\text{-}local}$ signals is dropped from the determination of T_H, due to an approach to saturation condition, then the same weighted average is performed, only with two weighting factors instead of three. Based on the applicant's response, it is clear that the weighting factors are not adjusted and there is no violation of 10 CFR Part 50, Appendix B, Criterion III. The applicant committed to update the TR for clarification. These changes were incorporated in a subsequent revision to the TR.

- There was concern with how a failure of the core inlet temperature signal (TC) would affect the protection system. Question 5 asks how the system responds to a "BAD" quality TC signal. The response states that an alarm is actuated in the MCR to notify operators to take appropriate action. The value of the signal (e.g., failed off-scale low, failed off-scale high or otherwise) will determine whether or not a trip is voted for. Based on the applicant's response, it is clear that a failure of a TC signal will result in appropriate operator action and, therefore, Question 5 is resolved.

- To determine how the margin to trip function interfaces with the protection system bi-stable controller, Question 6 asks how the margin to trip signal feeds into the logic that generates a division trip vote and if the margin to trip signal is used elsewhere. The response states that the margin to trip signal is directly input to the trip bi-stable controller, which looks for a negative value to allow for a trip vote. The signal also goes to the MCR for alarm and display. It is also stated that the margin to trip signal is hardwired into the PLS and that the information is available for use by other systems if needed. Based on clarification of how the margin to trip signal interfaces with the protection system, Question 6 is resolved.

- Question 7 refers to the use of time constants in the TH, TC, and OPΔT and OTΔT margin to trip signal development. Clarification was requested regarding the extra lag term and also how the values of these constants differ from those in the previously approved analog methodology. The response indicates that the extra lag term is dedicated to signal noise filtering and does not affect the shape of the output signal (i.e., it is comparable to the output signal that uses a first order lag term). The TR discusses the possible factors that go into calculation of the net lead and lag constants, many of which are optional, and furthermore are determined in the plant-specific safety analysis of record to verify the adequacy of the protection system. The responses to Questions 9-11 give typical values assumed in performed analog versus digital comparative analyses. Based on the provided clarification, Question 7 is resolved.

- Question 8 refers to the use of the bias coefficient and the conversion factor used in calculating the ΔT power signal. The question asks how the constants are determined and how often they must be adjusted. The response states that the bias coefficient will be adjusted so that the ΔT power signal indicates zero at hot zero power. This action is mandatory as part of the channel calibration required by TS surveillance requirement (SR) 3.3.1.9, which requires calibration every 24 months. The conversion factor is a gain adjustment that is adjusted as necessary in compliance with SR 3.3.1.3, which requires comparing the ΔT power signal to the calorimetric power, similar to the neutron flux power range signal surveillance, every 24 hours. Based on the provided clarification, Question 8 is resolved.

- Questions 9 through 11 ask the applicant to discuss the differences between the analog and digital based dynamic response when a trip occurs. Provided in the response are the assumed time constants used in comparing the analog response to the digital response. The differences are shown in a comparative example. Additional concern was expressed with regard to the Chapter 15 design basis accidents, which credit the OPΔT and OTΔT reactor trips. It was asked if the Chapter 15 accidents were revised to include the revised digital-based reactor trip functions.

  The comparisons between the two trip responses given in response to RAI-TR36-012 show that the trip responses are similar, which provides reasonable assurance that the dynamic compensation terms applied to the digital-based method and the proposed trip functions are appropriate. It was stated that the Chapter 15 design basis accidents were not updated to reflect the digital-based functions since the comparative studies performed confirmed that the digital-based method closely simulates the analog-based method without a loss of safety margin. In Revision 19 of the DCD, the applicant replaced the analysis results in DCD Section 15.4.2 with the re-analysis results described in response to RAI-TR36-012, which were based on the same cases previously analyzed. Various parts of DCD Section 15.4.2 are revised to reflect the revised analysis and results, as shown in Table 15.4-1 and Figures 15.4.2-1 through 15.4.2-15. The revised analysis used the upgraded digital-based OTΔT reactor trip function and NRC-approved methods, including the revised thermal design procedure (RTDP) and the LOFTRAN code. The revised analysis maintains the consequential loss of offsite power occurring 3 seconds after the turbine trip. Because the AP1000 PMS design is such that turbine trip occurs 5 seconds following a reactor trip condition being reached, the loss of offsite power and the resulting RCP coastdown occur 8 seconds after the reactor trip. Since the minimum DNBR occurs immediately after the reactor trip, the loss of offsite power has no effect on the minimum DNBR results. The revised results continue to show that the DNBR does not fall below the design limit DNBR. Therefore, fuel integrity and adequate fuel cooling are maintained. The peak reactor coolant system (RCS) pressure remains less than 110 percent of the design pressure. Therefore, the revised analysis continues to meet the acceptance criteria of NUREG-0800 Section 15.4.2. Questions 9 through 11 are, therefore, resolved.

- Question 12 was asked to resolve a discrepancy between time constants reported in APP-GW-GLR-137 and those shown in Figure 7.2-1 (Sheet 5) of the DCD. An updated figure was provided with the RAI responses for consistency, and Question 12 is therefore resolved. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- Question 13 asked the applicant to ensure that the TS and bases are consistent with the information provided in APP-GW-GLR-137. The response states that appropriate changes are being made. In Revision 19 to the AP1000 DCD, the applicant made an appropriate change to the DCD text, which resolves this issue.

- Section 5.0, "References," of APP-GW-GLR-137 needs to be updated since the currently referenced topical report discusses the methodology used to determine certain uncertainties that factor into the calculation of the OPΔT and OTΔT setpoints. APP-GW-GLR-137 states that a revision to the topical report will be issued at a later date. This change was incorporated in Revision 1 to the TR.

The staff concludes that, based on the information provided in APP-GW-GLR-137 and the responses to RAIs, no significant change was made to the functional output and underlying methodology for the digital based OPΔT and OTΔT margin to trip functions and it is, therefore, concluded that SAFDLs will not be exceeded. Improvements to the measurement of core power for input to the margin to trip calculation were made by using actual core parameters versus using differential temperature. This change addresses a source of previous inaccuracy in the trip functions. To provide further assurance that the ΔT power signal used in the margin to trip calculation is valid, the signal is also compared to the plant calorimetric heat balance routinely performed in SR 3.3.1.3 as given in Chapter 16 of the DCD.

The setpoint calculated in the digital-based methodology for the OTΔT setpoint is based on allowable core power as a function of pressurizer pressure and core inlet temperature instead of allowable ΔT as a function of T_{AVG}. This is a simpler method and allows for direct translation of the appropriate DNB thermal design limits into the OTΔT trip function. The OPΔT setpoint calculation is unchanged in the digital methodology, except for the units. The OPΔT setpoint is manually fixed at a determined power level and only changes as a function of the adverse axial offset.

7.A.3 Conclusion

After reviewing APP-GW-GLR-137, the staff finds the proposed OPΔT and OTΔT reactor trip function calculational methodology to be acceptable and considers Open Item OI-SRP16-CTSB-42 resolved.

8. ELECTRIC POWER SYSTEMS

8.2 Offsite Power Systems

8.2.2 Offsite Circuits within the AP1000 Scope of Design

In the second paragraph of Section 8.2.2 of NUREG-1793, "Final Safety Evaluation Report Related to Certification of the AP1000 Standard Design," the Nuclear Regulatory Commission (NRC) staff stated that the main generator normally provides power to the main alternating current (ac) power system. When the main generator is not available, the generator output breaker is opened and the plant auxiliary power comes from the switchyard by back feeding through the main step-up transformers and the unit auxiliary transformers (UATs). There is also a maintenance source provided through a reserve auxiliary transformer (RAT). The maintenance source is site-specific, and bus transfer to the maintenance source is manual. In Section 8.2.3.4, "Specific Interface Requirements for Supporting Chapter 15 Analyses," of NUREG-1793, the NRC staff stated that the AP1000 design uses no automatic transfers of reactor coolant pump (RCP) buses to alternate power supply. In addition, in Section 8.3.1, "Onsite Power Systems," of NUREG-1793, the NRC stated that the 6.9 kilovolt (kV) buses are provided with access to the maintenance source through normally open circuit breakers connecting the bus to the RAT and that bus transfer to the maintenance source is manual.

In technical report (TR) (TR-79), "Electrical System Design Changes," Revision 1, the applicant added a fast bus transfer scheme, along with an operator-initiated maintenance transfer, to the AP1000 design. This change required installation of an additional RAT to allow complete bus transfer from UATs to RATs.

As a result of the above changes to Tier 2, the corresponding portions of the Tier 1 AP1000 design control document (DCD) Table 2.6.1-3 (untitled) and Figure 2.6.1-1, "Main ac Power System," were affected.

8.2.2.1 Evaluation

The proposed change will allow transfer of 6.9 kV RCP buses from the UATs to the RATs. The applicant stated that the addition of the fast bus transfer scheme will avoid a reactor trip resulting from component failure or spurious actuation of the protective relaying associated with any of the main step-up transformers, UATs, or isophase bus duct, which would cause an RCP trip and a reactor trip. Thus, when either normal or preferred power supply is unavailable because of an electrical fault at the main step-up transformer, UAT, isophase bus duct, or nonsegregated bus duct, fast bus transfer will be initiated to transfer the loads to the RATs. In addition to the above, the applicant has also added operator-initiated, sync-supervised, closed-transition transfers on a bus-by-bus basis to the AP1000 design.

The staff was concerned about a statement in Section 8.3.1.1.1 of the AP1000 DCD, which reads:

> ...in the event of a loss of voltage on these buses, the diesel generators are automatically started and connected to the respective buses and in the event of a fast bus transfer, the diesel generator connection to the bus is delayed such that the fast bus transfer is allowed to initiate.

This statement implies that the diesel generator is already running during the fast bus transfer and its connection to the bus is delayed. In request for additional information (RAI) RAI-SRP8.3.1-EEB-02, the NRC requested that the applicant clarify when the diesel generator would start during fast bus transfer. In its response dated July 11, 2008, the applicant revised Section 8.3.1.1.1 of the DCD to clarify the statement as follows:

> In the event where a fast bus transfer initiates but fails to complete, the diesel generator will start on an undervoltage signal, but if a successful residual voltage transfer occurs, the diesel generator will not be connected to the bus, as the successful residual voltage transfer will provide power to the bus prior to the diesel connection time of 2 minutes.

The staff concluded that this revision to the AP1000 DCD satisfies its concern. The staff verified that Revision 19 to the AP1000 DCD includes the foregoing change.

8.2.2.2 Conclusion

The staff has reviewed these changes and concludes that the additions of a RAT and the bus transfer scheme to the AP1000 design provide additional plant availability and enhance the offsite power supply to the safety-related battery chargers, RCPs, and those priority loads provided for defense-in-depth functions.

8.3 Onsite Power System

8.3.1 AC Onsite Power System

8.3.1.1 Electric Circuit Protection

In this section of NUREG-1793, the NRC discussed the ratings and major types of protection systems employed for the AP1000 medium voltage switchgear. In TR-79, the applicant made the changes described in the sections below.

8.3.1.1.1 Rating of 6.9-kV Switchgear Buses

In TR-79, the applicant proposed to revise the short-circuit rating of 6.9-kV switchgear buses from the current 500 megavolt amps (MVA) (40 kilo-amperes (kA)) to 63 kA. The Tier 2 portions of the AP1000 DCD affected in the electrical area include Figure 8.3.1-1, "AC Power Station One Line Diagram," and Table 8.3.1-3, "Component Data – Main Power System."

8.3.1.1.1.1 Evaluation

Originally, the AP1000 DCD included the short-circuit rating of the 6.9-kV switchgear buses as 500 MVA (approximately 40 kA). The applicant proposed to revise the short-circuit rating from the current 40 kA to 63 kA. The applicant stated that the value change from 40 kA to 63 kA is based on revised short-circuit calculations that demonstrate that 63 kA is bounding, given the UAT/RAT size and the expected largest motor size driving the allowed impedance of the UAT/RAT transformers. In RAI-SRP8.3-EEB-01, the staff asked the applicant to justify this change in terms of the reason the interrupting rating changed from 40 kA to 63 kA and whether the proposed change affects the onsite distribution system analysis.

In its response dated October 17, 2008, the applicant stated that the onsite distribution system analysis supports the described engineering values. The value change from 40 kA to 63 kA is based on a computation of short-circuit current from an infinite source upstream of a UAT/RAT, neglecting the minimal contribution between the X-Y secondary windings of the transformer. In addition, the applicant considered a conservative assumption of a 100-percent motor load on a 100-percent loaded transformer winding using a 6.5 multiplier for motor short-circuit contribution. This value was computed while establishing a transformer impedance low enough to allow for starting the single largest motor. This computation demonstrates that 40 kA is inadequate and that 63 kA is bounding given the UAT/RAT size and the expected largest motor size driving the allowed impedance of the UATs/RATs.

8.3.1.1.1.2 Conclusion

The NRC reviewed this change and concludes that the proposed rating of 63 kA is acceptable because, in the current design, the starting current value of the largest motor is 58 kA, which is well in excess of 43 kA for the 6.9-kV switchgear bus rating. Therefore, a 63-kA rating for the 6.9-kV switchgear bus bounds the largest motor. Based on this discussion, the issue is resolved.

8.3.1.1.2 Air Cooled Chillers

In TR-79, the applicant proposed to revise AP1000 DCD Figure 8.3.1-1 to show the air-cooled chillers VWS-MS-02 and VWS-MS-03 being fed from the 6.9-kV buses ES-1 and ES-2, respectively.

The Tier 2 portions of the AP1000 DCD affected include Figure 8.3.1-1.

8.3.1.1.2.1 Evaluation

Figure 8.3.1-1 of the AP1000 DCD did not show the air-cooled chillers VWS-MS-02 and VWS-MS-03 being fed from the 6.9-kV buses ES-1 and ES-2, respectively. These loads were previously connected at the 480-volt (V) level. The applicant revised the DCD to reflect the connection of these loads directly to the 6.9-kV buses.

8.3.1.1.2.2 Conclusion

The staff reviewed this change and concludes that these loads were erroneously shown connected at the 480-V level and that the connection of the air chillers to the 6.9-kV buses is consistent with the design for this size of motor load. Therefore, the proposed change is acceptable.

8.3.1.1.3 Raw Water Feeder Breaker Change

In TR-79, the applicant proposed to revise AP1000 DCD Figure 8.3.1-1 to show the three raw water pumps and their auxiliaries. The Tier 2 portions of the AP1000 DCD affected include Figure 8.3.1-1.

8.3.1.1.3.1 Evaluation

AP1000 DCD Figure 8.3.1-1 in Revision 15 showed that three feeders for three raw water pumps were fed directly from 6.9-kV buses and their auxiliaries were powered from 480-V

buses. The applicant has proposed to have each of these three feeders support the respective raw water pump and the associated auxiliaries of that pump. The proposed change allows a single feeder to the raw water pump house and distributes power within that structure.

8.3.1.1.3.2 Conclusion

The staff reviewed this change and concludes that it has no impact on the safety systems; therefore, it is acceptable.

8.3.1.2 Standby Diesel Generators

Section 8.3.1.2, "Standby Diesel Generators," includes the staff's review of generator exciter and voltage regulator systems as well as power sources.

8.3.1.2.1 Generator Exciter and Voltage Regulator Systems

In the sixth paragraph of Section 8.3.1.2 of NUREG-1793, the staff stated that the generator exciter and voltage regulator systems are capable of providing full voltage control during operating conditions, including postulated fault conditions.

In TR-79, the applicant proposed to delete the word "static" from the description of the exciter type so that a more readily available design may be used.

In addition, the applicant revised the nominal power ratings of various pieces of equipment and diesel generator loading in Table 8.3.1-2, "Onsite Standby Diesel Generator ZOS MG 02B Nominal Loads," of the AP1000 DCD.

8.3.1.2.1.1 Evaluation

There are no regulatory criteria that require a specific diesel generator exciter type. Revision 15 of the AP1000 DCD specified that each onsite diesel generator has a "static" exciter. NUREG-1793 did not identify what type of exciter was used for standby diesel generators. The staff concludes that deleting the word "static" from the description of the exciter type will provide applicant flexibility in choosing what equipment is to be procured and there is no impact on performance requirements.

In Revisions 16 and 17 of the AP1000 DCD, the applicant revised the rating and operating load sizes of various pieces of equipment and updated the affected diesel loading tables. The staff reviewed the revised loads in Tables 8.3.1-1 and 8.3.1-2 and found that the total loads exceed the rating of diesel generators. The portions of the AP1000 DCD affected include Tables 8.3.1-1 and 8.3.1-2. In RAI-SRP8.3.1-EEB-01, the staff noted that the sum of the total loads (automatic and manual) listed in each revised table exceeds the continuous rating of each diesel generator and requested the applicant to justify why it is acceptable to exceed the continuous rating of the diesel generator at this stage of the design. Also, the applicant was requested to describe provisions included in the design that will prevent overloading of the diesel generators when manual loads are powered from the diesel generator.

In its response dated June 23, 2009, the applicant stated that onsite standby diesel generators have a nominal rating of 4000 kilowatt (kW); however, the units will accept loads up to the overload ratings of the diesel generators for the period of time specified for those ratings. The intent of the diesel generator loading is to accept all automatic loads followed by other loads to

be manually added under operator control. In addition, the applicant stated that the abnormal operating procedures include diesel generator load management details to be followed after the automatic load sequencing. These procedures will identify that additional loads can be manually loaded at the operator's option. The operator will assess plant conditions and available diesel generator capacity to determine if these additional components should be started. The operator has main control room indication of the current power demand on each of the diesel generators upon which to base his decision. The staff finds the response to the RAI to be acceptable.

8.3.1.2.1.2 Conclusion

The staff concludes that removing the word "static" from the type of exciter would provide the applicant more flexibility in choosing other excitation systems; therefore, the proposed change is acceptable. The staff also finds the changes made to Tables 8.3.1-1 and 8.3.1-2 for auto-connected loads to be acceptable because: 1) the total auto-connected loads (2706 kW, 3126 kW) on each diesel generator is still within the continuous rating of 4000 kW; and 2) the procedures that will prevent overloading of the diesel generators when manual loads are powered from the diesel generator are included.

8.3.1.2.2 Power Sources

In the third paragraph of Section 8.3.1.2, "Standby Diesel Generators" of NUREG-1793, the staff stated that during plant startup, shutdown, and maintenance, the generator breaker is opened. Under this condition, the preferred power supply system provides the main ac power from the high-voltage switchyard through the main step-up transformers and two UATs. Each UAT supplies power to about 50 percent of the plant loads. The UATs have two identically rated 6.9 kV secondary windings.

In TR-114, "AP1000 Auxiliary Building Boiler Sizing and Design," the applicant added a third two-winding UAT sized to accommodate the electric auxiliary steam boiler and site-specific loads. In addition, in Revision 17 to the AP1000 DCD, the applicant revised Section 8.3.1.1.1, "Onsite AC Power System," to describe the neutral overcurrent protection for the RATs.

As a result of the above changes to Tier 2, the portions of the Tier 1 AP1000 DCD affected include Figure 2.6.1-1, "Main AC Power System," and Tables 2.6.1-3 and 2.6.1-5.

8.3.1.2.2.1 Evaluation

The applicant has proposed a design change from a diesel-fired auxiliary steam boiler to an electric auxiliary steam boiler for the AP1000 to alleviate issues in current plants related to operational problems caused by fuel fouling in diesel-fired boilers in standby service. This change reduces the size requirement on the auxiliary boiler by over 50 percent (from approximately 70 megawatts (MW) to 25 MW). As a result, the applicant added a third two-winding UAT sized to accommodate the electric boiler and site-specific loads. The third UAT would be located outside the turbine building in the transformer area. A 25-MW electric boiler would be installed in the boiler room of the turbine building along with its associated switchgear ES7, load center, and motor control center. This design change would not have any impact on the safety systems.

The staff was concerned that the applicant did not provide any neutral overcurrent protection for the RATs. In RAI-SRP8.2-EEB-03, dated March 13, 2008, the staff asked the applicant to

justify its failure to provide neutral overcurrent protection for the RATs. In its response dated April 22, 2008, the applicant stated that it had inadvertently omitted the neutral overcurrent protection from page 8.3-2 of the AP1000 DCD. The applicant committed to modify DCD Section 8.3.1.1.1 to show neutral overcurrent protection for the RATs. The staff finds the above design to be consistent with the recommendations of Institute of Electrical and Electronics Engineers (IEEE) Standard 666, "IEEE Design Guide for Electric Power Service Systems for Generating Systems."

The staff confirmed that Revision 19 to the AP1000 DCD includes the neutral overcurrent protection for the RATs.

8.3.1.2.2.2 Conclusion

The staff reviewed this change and concludes that the addition of the third UAT and its associated switchgear ES7 has no impact on the safety systems and, therefore, the proposed change is acceptable. In addition, the staff finds the inclusion of the overcurrent protection for the RATs acceptable because it is consistent with expected engineering practice.

8.3.1.3 Ancillary AC Diesel Generators

In the first paragraph of Section 8.3.1.2 of NUREG-1793, the staff stated that the applicant has included two ancillary diesel generators located in the annex building to provide power to meet the post-72-hour power requirements following an extended loss of offsite power sources. Each ancillary diesel generator output is connected to a distribution panel.

In TR-79, the applicant proposed to revise Figure 8.3.1-3 of AP1000 DCD Tier 2 to reflect a four-wire 100 amp distribution panel from a three-wire 50 amp distribution panel, and a 100 amp breaker for both the diesel generator to the bus and for the test load tie to the bus from the 50 amp breaker.

8.3.1.3.1 Evaluation

There are no regulatory requirements concerning the number or rating of wires to or from a nonsafety-related ancillary diesel generator. The staff reviewed this change to determine whether it would adversely affect the design. Currently, AP1000 DCD Figure 8.3.1-3 shows the size of the ancillary diesel generator distribution panels as 50 amp with an incoming breaker of 30 amp from the generator. The applicant proposed to revise Figure 8.3.1-3 of the DCD to reflect a four-wire 100 amp distribution panel from a three-wire 50 amp distribution panel, and a 100 amp breaker for both the diesel generator to the bus and for the test load tie to the bus from the 50 amp breaker. The applicant stated that since the full load current of the generator is 53 amps, the main breaker of the distribution panel should be sized at the full capacity of the generator at a minimum. The diesel generator test load will also be changed to 100 amp to allow for this generator to be tested at full capacity. The applicant made its selection based on the 480 volts alternating current (Vac) standard-sized distribution panels available in the industry. In addition, to facilitate the use of this source as a feed to 277 Vac lighting circuits, these panels would be changed from three-wire system to four-wire system.

8.3.1.3.2 Conclusion

The staff reviewed these changes and concludes that the original rating of the bus and breakers of the panel was undersized and that the proposed revised rating of the diesel generator

distribution panel is adequate because it exceeds the full load current of the diesel generator. Therefore, the proposed change is acceptable.

8.3.2 Direct Current Power and Uninterruptible Power Systems

8.3.2.1.1 Class 1E dc Distribution

In Section 8.3.2.1.1 of NUREG-1793, the staff stated that the Class 1E direct current (dc) power system consists of four independent 125 V Class 1E dc safety system divisions (Divisions A, B, C, and D). Divisions A and D are each comprised of one battery bank, one switchboard, and one battery charger. Divisions B and C are each comprised of two battery banks, two switchboards, and two battery chargers; however, in Revision 17 of the AP1000 DCD, the applicant changed the system voltage for the operation of Class 1E dc loads from 125 volts direct current (Vdc) to 250 Vdc.

The portions of the AP1000 DCD affected include pages 8.1-2, 8.1-3, 8.3-10, 8.3-14 through 8.3-17, 8.3-22 through 8.3-26, and Tables 8.3.2-1 through 8.3.2-7. The portions of the Tier 1 AP1000 DCD affected include Section 2.6.3, "Class 1E dc and Uninterruptable Power Supply System," Table 2.6.3-1 (untitled), Table 2.6.3-3, "Inspections, Tests, Analyses and Acceptance Criteria," and Table 2.6.3-4 (untitled).

As part of the staff's review of the proposed changes to the Class 1E dc system, the staff issued several RAIs. The staff's evaluation of the applicant's responses to the RAIs is as follows.

8.3.2.1.1.1 Evaluation

In RAI-SRP8.3.2-EEB-01, the staff requested that the applicant provide a discussion as to how this voltage change would impact motors, cables, protective devices, switchboard, and other equipment, as applicable. Also, the applicant was asked to describe how motor sizing and cable sizing would still be compatible with valve loads.

In its response dated May 7, 2009, the applicant stated that dc motor-operated valve motors would draw less current to accomplish the required power. Cable sizes would be reduced considerably. As the current drawn by the same size (kW) motor is halved and the total voltage drop allowed is doubled, the cable sizes would be reduced accordingly. Electrical distribution equipment would require, nominally, one half the current rating. The staff agrees with the applicant that by increasing the system voltage the current would be reduced, the cable sizes would be reduced, and the electrical distribution equipment would require less current. Therefore, the staff finds applicant's response to be acceptable and finds this issue resolved.

In RAI-SRP8.3.2-EEB-02, the staff noted that AP1000 DCD Section 8.3.2.1 indicates that the operating voltage range of the Class 1E batteries is 210 to 280 Vdc and the maximum equalizing charge voltage for the Class 1E batteries is 280 Vdc. The applicant was asked to confirm that the connected dc equipment is designed to operate up to the maximum voltage 280 Vdc.

In its response dated May 7, 2009, the applicant stated that all connected equipment design specifications would include the new voltage limit requirements. Based on the above, the staff finds that since specification of each piece of equipment would include the revised voltage specification, the applicant's response is acceptable and this issue is resolved.

In RAI-SRP8.3.2-EEB-03, the staff requested that the applicant provide the load profiles (duty cycle) from one minute to 24/72 hours for each of the 24-hour and the 72-hour Class 1E 250 Vdc batteries. The applicant was asked to discuss battery margins (aging margin, design margin, temperature correction factor, margin associated with float current for 100 percent state of charge) and the expected service life of these batteries.

In its response dated June 23, 2009, the applicant stated that for battery aging margin, a factor of 25 percent would be used for a 20 year qualified battery. Temperature correction would be based on minimum temperature of 16° Celsius (C) (60° F). With regard to float current margin, the applicant stated that this margin is described as a consideration for quick turnaround to service after discharge. Since the electrical design described in the AP1000 DCD utilizes a spare battery that can replace any safety-related battery, there is no immediate need to replace the discharged battery. The replacement interval/service life of the batteries will be in accordance with the testing program replacement requirements. Replacement intervals will be based on degraded performance in accordance with the required test program. However, the applicant did not provide the load profiles for 24-hour and 72-hour batteries as requested by the staff. The staff identified this as Open Item OI-SRP8.3.2-EEB-03 in the safety evaluation report (SER) with open items.

Subsequently, in a letter dated May 10, 2010, the applicant stated that the nominal loads on the batteries are identified in AP1000 DCD Tables 8.3.2-1, 8.3.2-2, 8.3.2-3, and 8.2.3-4 and that the design is based on intelligent assumptions on the loads. Also, as part of the response to the above open item, the applicant provided document, APP-IDS-EOC-001, Revision 0, "Class 1E 250 V DC Battery Sizing, Charger Sizing and Available Short Circuit Current," for the staff's review to assess the adequacy of the 24-hour and 72-hour batteries. The staff reviewed the applicant's onsite documentation which included the load profiles for loss of offsite power and loss-of-coolant accident (LOCA) from one minute to 24/72 hours for each of the 24-hour and the 72-hour Class 1E 250 Vdc batteries in this report. Based on its review, the staff concluded that since the AP1000 Class 1E 250 Vdc batteries are sized in accordance with the recommendations of IEEE Standard 485, "IEEE Recommended Practice for Sizing Lead-Acid Batteries for Stationary Applications," (which provides methods for defining the dc load and for sizing a battery to supply that load for stationary batteries), there is reasonable assurance that the batteries would be designed to have adequate capacity to meet their respective load profile. In addition, the staff determined that the battery qualification program and the applicable surveillance requirements in accordance with plant technical specifications would ensure that the batteries would envelop their designed load profiles throughout their designed life. On this basis, the staff considers Open Item OI-SRP8.3.2-EEB-03 resolved.

In RAI-SRP8.3.2-EEB-04, the staff requested that the applicant describe how the 24-hour and the 72-hour 250 Vdc batteries would be qualified for service life: If safety-related batteries would be qualified using the recommendations of IEEE Standard 535, "Standard for Qualification of Class 1E Lead Storage Batteries for Nuclear Power Generating Stations," it is not clear how the standard applies since the standard was written under the assumption of an 8-hour duty cycle. Since AP1000 design duty cycles are significantly longer than 8-hour duty cycle and IEEE Standard 535 does not apply to duty cycles longer than 8 hours, the applicant was asked to describe how these batteries would be qualified for extended duty cycles of 24 hours and 72 hours. The applicant was also asked to discuss the failure mode(s) for both the 24-hour and 72-hour duty cycle batteries.

In its response dated May 7, 2009, the applicant stated that it intends to qualify the AP1000 safety-related batteries for 24-hour and 72-hour duty cycles through the implementation of

industry standards IEEE Standard 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations"; IEEE Standard 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"; and IEEE Standard 535-1986 as they apply to the equipment.

The qualification process for the AP1000 24-hour and 72-hour duty cycle batteries would be outlined in a test plan. Qualification of the batteries would be accomplished by type testing of both duty cycle designs to the AP1000 service conditions associated with their projected service life. In the qualification process, the batteries would be subjected to aging (thermal, wear/operational), abnormal environmental, and seismic conditions. There are no radiation and normal vibration conditions associated with the mounting locations of the batteries. Aging under normal and abnormal service conditions would be performed to degrade batteries to their end-of-life such that the safety function after the design basis event (DBE) (seismic) would be verified.

The aging conditions would include both electrical (chemical) cycling and thermal accelerated aging. Electrical (chemical) cycling would be performed in compliance with IEEE Standard 323. The proposed electrical (chemical) cycling is in line with Section 8.2.2 (6) of IEEE Standard 535-1986 for cases when the service conditions are more severe than those specified in the standard. The electrical (chemical) cycling of the batteries is based on the AP1000 maintenance/surveillance requirements with no less than 10 percent margin. During the testing process, the service and performance tests would be performed in conjunction with the thermal accelerated aging test of the batteries to place the batteries in an end-of-life condition. Upon completion of the battery aging, abnormal environmental testing to the AP1000 mild environment abnormal conditions would be performed. Following the abnormal environmental testing, seismic testing and a hard rock high frequency screening test would be performed.

At the completion of seismic testing, a post-seismic battery service test would be performed. The service test is used to demonstrate equipment functionality during and after the DBE (seismic), which is a requirement in accordance with IEEE Standard 344 and IEEE Standard 323. This is different from IEEE Standard 535, which only requires a performance test to be performed. In the process of performing the qualification testing of the AP1000 batteries, the program would identify any failure mechanisms that may surface during the projected service life in an AP1000 plant.

In a May 21, 2009 conference call, the staff requested that the applicant provide its step-by-step, detailed qualification test plan showing testing for desired qualified life of the batteries. However, the applicant did not provide its qualification test plan for the batteries prior to issuance of the SER with open items. This issue was tracked as Open Item OI-SRP8.3.2-EEB-04 in the SER with open items.

Subsequently, in a letter dated March 2, 2010, as part of the response to the above open item, the applicant provided document, EQ-TP-59-APP (APP-DB01-VPH-001), Revision 0, "AP1000 Test Plan for Safety Related 250 Vdc Batteries," for the staff's review. The staff reviewed the applicant's onsite documentation supporting the qualification methodology for the 24-hour and 72-hour extended duty cycle batteries. The applicant provided detailed steps that would be followed to qualify the batteries as requested by the staff. The qualification would be based on the requirements of IEEE Standard 323-1974, IEEE Standard 344-1987, and IEEE Standard 535-1986. Qualification of the Class 1E batteries would be performed by testing. Due to the difference in duty cycle, the test sequence would be performed on two groups of test cells. One group would be cycled and tested to the 24-hour duty cycle for AP1000 and the

other group would be cycled and tested to the 72-hour duty cycle for AP1000. The test plan includes a series of modified performance tests at two year intervals that envelop the load profile of a service test throughout the installed 20-year life of the batteries.

In a public meeting held on May 27, 2010, to discuss the AP1000 Chapter 8 open items, the staff informed the applicant that its qualification test plan for the batteries is reasonable, but that the applicant must capture the qualification test plan as part of its licensing basis.

In a letter dated June 18, 2010, the applicant stated that it would revise its DCD to include the battery qualification test program. The applicant included the proposed revised Section 8.3.2.1.4, "Description," of the DCD as part of its response as follows:

> The qualification test program for AP1000 24-hour and 72-hour class 1E batteries meets or exceeds the requirements of IEEE Std 323, IEEE Std 344 and IEEE Std 535 including required and recommended margins and is in regulatory compliance with RGs 1.89, 1.100 and 1.158. The test program requires that the battery be subjected to accelerated thermal aging and discharge cycling (wear aging) in accordance with Institute of Electrical and Electronics Engineers (IEEE) Std 323 and IEEE Std 535 over its qualified life objective followed by the DBE seismic event performed in accordance with IEEE Std 344. In addition, following the aging process, the test specimens shall be subjected to environmental testing to verify the equipment's ability to operate in postulated abnormal environmental conditions during plant operation. Discharge cycling will be performed as a potential aging mechanism prior to seismic testing using Type 3 modified performance test method in accordance with IEEE Std 450-2002 at intervals representative of the AP1000 surveillance test requirements of the batteries with 10 percent margin in the number of discharge cycles which establishes margin for the expected life of the battery. Thus, magnitude / duration (modified performance test versus service and performance tests) and test interval envelop the AP1000 and industry cycling requirements. If new battery failure modes are detected during the qualification testing, these failure modes will be evaluated for any potential changes to the technical specification's surveillance requirements and revision to maintenance procedures required to ensure identification of degradation prior to reaching those failure modes during plant operation. Following the qualification process, a report that uniquely describes step-by-step the tests performed and results and addressing any deficiencies and repairs including photographs, drawings, and other materials will be maintained for records.

Based on the above, the staff concluded the applicant's test plan provided in EQ-TP-59-APP (APP-DB01-VPH-001) satisfies the recommendations of IEEE Standard 323-1974, IEEE Standard 344-1987, and IEEE Standard 535-1986 and provides reasonable assurance that its batteries and racks will perform their required functions throughout their qualified life. Therefore, Open Item OI-SRP8.3.2-EEB-04 is resolved subject to the verification that the AP1000 DCD is updated to include the revised paragraph. The staff confirmed that Revision 19 to the AP1000 DCD includes the revised paragraph.

In RAI-SRP8.3.2-EEB-05, the staff stated that in order to assess the adequacy of the dc power systems, it needed the results of 250 Vdc battery and battery charger sizing calculations, battery terminal voltage calculations, short circuit calculations, voltage drop calculations, and the associated assumptions used.

In its response dated May 7, 2009, the applicant stated that the results of 250 Vdc battery and battery charger sizing calculations, battery terminal voltage calculations, short circuit calculations, and voltage drop calculations would be available during a design review stage. This was tracked as Open Item OI-SRP8.3.2-EEB-05 in the SER with open items.

Subsequently, in a letter dated April 21, 2010, as part of the response to Open Item OI-SRP8.3.2-EEB-05, the applicant provided document, APP-IDS-EOC-001, for the staff's review. The document includes information to assess the adequacy of the battery banks and chargers for use in the Class 1E dc and uninterruptable power supply (UPS) for the AP1000 plant. The staff reviewed the applicant's onsite documentation that describes applicant's methodology for sizing batteries, chargers, and the available short circuit current from these sources. The staff verified that AP1000 Class 1E batteries are sized in accordance with the recommendations of IEEE Standard 485, and the battery chargers are sized in accordance with the recommendations of IEEE Standard 946, "IEEE Recommended Practice for the Design of DC Auxiliary Power Systems for Generating Stations." These standards provide guidance for sizing batteries and battery chargers. During its review of the document, the staff noticed that the required capacity of an IDSC-DB Division C, 72-hour battery is 2430 AH, while the batteries are rated for only 2400 AH in accordance with the AP1000 DCD. In a public meeting held on May 27, 2010, the staff asked the applicant to justify the apparent difference in the required capacity of the 72-hour battery per calculation versus the stated capacity of the 72-hour battery listed in Table 8.3.2-5 of the DCD.

In a letter dated June 18, 2010, the applicant stated that the required capacity of the IDSC-DB Division C, 72-hour battery in APP-IDS-EOC-001, Revision 0, would be revised to be made consistent with the AP1000 DCD. The staff finds the applicant's commitment acceptable because the revised calculations will ultimately be reviewed by the staff as part of the inspections, tests, analyses, and acceptance criteria (ITAAC) for the dc system, as described in AP1000 DCD Tier 1, Table 2.6.3-3, "Inspections, Tests, Analyses and Acceptance Criteria."

Based on its review, the staff concluded that the applicant's methodology for sizing batteries and battery chargers using these standards provides reasonable assurance that the batteries and battery chargers will be sized adequately and perform their safety functions as designed. The staff also verified that the dc switchboard rating exceeds the available short circuit current contributions from the batteries, battery chargers and regulating transformers. Further, ITAAC verifying that the batteries, chargers, and distribution systems are adequately designed are identified in AP1000 DCD Tier 1, Table 2.6.3-3. The staff will ultimately verify these ITAAC to ensure that the dc distribution system components, including the batteries and battery chargers, are adequately designed and the as-built design conforms to the approved plant design and applicable regulations. Therefore, Open Item OI-SRP8.3.2-EEB-05 is resolved.

In RAI-SRP8.3.2-EEB-06, the staff noted that AP1000 DCD Section 8.3.2.1.1.1, for 72-hour 250 Vdc batteries states, "Each switchboard connected with a 72-hour battery bank supplies power to an inverter. No load shedding or load management program is needed to maintain power during the required 24-hour safety actuation period." The staff requested that the applicant clarify if manual actions would be necessary to maintain power during the required 72 hours and to describe the loads that would be shed after 24 hours.

In its response dated May 7, 2009, the applicant confirmed that no operator action is necessary during the 72-hour period to maintain the adequacy of either the 24-hour or 72-hour portions of the dc power system. This satisfies the staff's concern and this item is resolved.

In RAI-SRP8.3.2-EEB-08, the staff noted that Figure 2.6.1-1, Tier 1 of the AP1000 DCD shows that the motor control centers that feed the safety-related 250 Vdc battery chargers are fed from 480 V load centers. The applicant was asked to provide a detailed drawing of the 480 V load centers, the motor control center (MCC) that feeds the battery chargers, and the dc MCC showing typical loads powered from these buses. The applicant was asked to describe how the 480 V load centers are protected from degraded voltage and frequency conditions and to provide the following information:

a. The results of an analysis of the onsite power distribution system to demonstrate that adequate voltages at terminals of the battery chargers are optimized for the maximum and minimum voltage variations of the offsite power for events such as a unit trip, LOCA, startup or shutdown.

b. A description of the analytical techniques, methodology, and assumptions used in performing the analyses. Also, provide the results of these analyses for each level of onsite electrical power distribution.

c. Identification of the analytical software (and its version) used for performing these studies and make available to the staff an electronic copy of the electrical distribution system model that forms the basis of the analytical studies.

In its response dated June 23, 2009, the applicant stated that the level of design detail requested would be available following the completion of the design review stage of the system. In general, the staff was seeking understanding of the applicant's approach to assuring consistency in the transfer to the combined license (COL) applicant: 1) analysis, calculations and assumptions made for maintaining adequate voltage regulation at safety-related equipment terminals; 2) the analysis and assumptions used to evaluate acceptable rating for equipment such as circuit breakers; and 3) the studies, acceptance criteria, and assumptions used to determine equipment sizing. This was tracked as Open Item OI-SRP8.3.2-EEB-08 in the SER with open items.

Subsequently, in letters dated February 1, 2010, and May 11, 2010, the applicant stated that to assure consistency in the transfer of design information to a COL applicant, it provides a COL applicant with a configuration-controlled model developed through the use of an Electrical Transient Analysis Program. The program includes the nonsafety-related ac design calculations, including design inputs, assumptions, methodologies, and acceptance criteria used in the development of the sizing basis, settings, load flow, short circuit, and voltage regulation. In addition, the applicant stated that it has performed an analysis of its onsite ac distribution system and that the results of the analysis of the onsite power distribution system demonstrate that adequate voltages at the terminal of the battery chargers are optimized for the maximum and minimum voltage variations of the offsite power for events, such as a unit trip, LOCA, startup or shutdown. The above will ensure that adequate voltages at terminals of the battery chargers are optimized for the maximum and minimum voltage variations of the offsite power to satisfy the requirements of 10 CFR Part 50, "Domestic licensing of production and utilization facilities," Appendix A, "General Design Criteria for Nuclear Power Plants," General Design Criteria (GDC) 17, "Electric Power Systems," with respect to their capacity and capability to perform their safety function. Although the applicant did not submit this information for staff review, the staff determined this response to be acceptable because ultimately all COL applicants would have to complete testing of the onsite (ac and dc) and offsite power systems, which are fully interconnected, to verify that the non-Class 1E ac power system will have an

acceptable design to support the safety-related loads. Therefore, Open Item OI-SRP8.3.2-EEB-08 is resolved.

In RAI-SRP8.3.2-EEB-09, the staff noted that the AP1000 is designed to sustain a load rejection from 100 percent power with the turbine generator continuing stable operation while supplying the plant house loads. The staff is concerned about the transient conditions where a significant voltage spike during islanding could cause high dc voltage conditions on the output side of the battery chargers. Operating experience (see NRC Information Notice (IN) 2006-18, "Significant Loss of Safety-Related Electrical Power at Forsmark, Unit 1, in Sweden" dated August 17, 2006) has shown that the voltage spike due to malfunction of the main generator exciter or during islanding could go as high as 130 percent, which could go undetected by normally-provided relaying and could cause damage to the safety-related equipment or mis-operation. In this regard, the applicant was asked to describe how the protective features of the inverter and the new battery chargers would be coordinated so that any voltage transient would not result in inadvertent loss of the inverters or the batteries.

In a letter dated June 23, 2009, the applicant stated that the battery charger input circuit will conduct power to charge the batteries when ac power is available. The battery charger is specified to return to operation after voltage drifts outside of an acceptable input voltage range. The battery charger is also a qualified isolation device, isolating the battery and the inverter from the nonsafety-related ac system. During the period where the battery charger is not conducting, the battery will carry the load. In addition, the applicant stated that it has considered over-voltage events with the potential to have effects upon plant safety-related equipment as provided under the direction of IN 2006-18. However, the applicant did not provide the details of how to avoid this kind of event in AP1000 design or identification of potential vulnerabilities and actions that could reduce the challenges for the control room operators. This potential event is significant in that it can cause the common mode failure in all four trains and, therefore, could result in the loss of all four trains of safety-related ac and dc power. Transient voltages on the ac input to the battery chargers can result in high dc voltages that could lead to failures of critical electrical and electronic components including electrical inverters unless they are properly protected. During such a voltage transient, the inverter voltage surge protection could trip before actuation of the battery charger protection if the battery charger and inverter dc input voltage protection settings are very close to each other. Therefore, it is necessary that the safety-related battery chargers and inverter trips be coordinated such that the associated inverters do not trip on during voltage transients on the ac distribution system. This was tracked as Open Item OI-SRP8.3.2-EEB-09 in the SER with open items.

Subsequently, in letters dated January 26, 2010, and May 11, 2010, the applicant stated that as part of the component design specification, the battery charger/inverter would be designed specifically with consideration of the Forsmark incident identified in IN 2006-18. Industry evaluations of this incident identify the lack of coordination as a primary causative issue. In addition, the applicant stated that the protective devices will be set so that the battery charger will not trip on the over-voltage resulting from load rejection and will be set low enough to protect the equipment. The inverter dc input protection will be set at least 10 percent higher than the battery charger output dc protection to prevent the inverter tripping before the battery charger.

In a public meeting held on May 27, 2010, to discuss AP1000 Chapter 8 open items, the staff informed the applicant that its response to the above open item was inadequate. The staff stated that the safety-related inverter high dc input voltage trip set point and the associated battery charger high dc output voltage trip set point should be coordinated in both magnitude

and time. The staff stated that the applicant should amend the DCD to include its response as modified.

In a letter dated June 18, 2010, the applicant provided a proposed revision to Section 8.3.2.1.4, "Maintenance and Testing," of the AP1000 DCD as part of its response to Open Item OI-SRP8.3.2-EEB-09 as follows:

> The inverter DC input protection will be set at least 10 percent higher than the battery charger trip setpoints to prevent the inverter tripping before the battery charger. The time delay for the inverter high dc input voltage trip will be set higher than the delay time delay for the battery charger to prevent the inverter tripping before the battery charger.

In addition, the applicant stated in its response dated May 11, 2010, that the battery charger function in the AP1000 design is to provide isolation between input ac and the dc system and to provide dc power when ac power is available. The staff noted that Section 8.3.2.2 of the AP1000 DCD states that Class 1E battery chargers and Class 1E voltage-regulating transformers are designed to limit the input (ac) current to an acceptable value under faulted conditions on the output side. Both have built-in circuit breakers at the input and output sides for protection and isolation. The circuit breakers are coordinated and periodically tested to verify their current-limiting characteristics. In the public meeting held on May 27, 2010, the staff requested the applicant explain how the requirement for periodic testing of the Class 1E battery chargers and Class 1E voltage-regulating transformers used as isolation devices would be satisfied by each COL applicant. The staff requested that the applicant indicate where this requirement would be located so that periodic testing of these devices was performed by each applicant to satisfy the recommendations of IEEE Standard 384, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," endorsed by Regulatory Guide (RG) 1.75, "Physical Independence of Electric Systems."

In the letter dated June 18, 2010, the applicant stated that a COL information item would be added to the AP1000 DCD to ensure that periodic testing was performed on the battery chargers and the regulating transformers. The applicant included the proposed revised Section 8.3.3, "Combined License Information for Onsite Electric Power," of the DCD as part of its response as follows:

> Combined License applicants referencing the AP1000 certified design will ensure that periodic testing is performed on the battery chargers and voltage regulating transformers.

Based on the above, the staff concluded that the applicant's modified response on battery charger and inverter trip setpoints satisfies the requirements of GDC 17 with respect to the capability of dc systems to perform their safety function. The staff determined that the addition of the above COL information item to the AP1000 DCD will ensure that periodic testing is performed on the battery chargers and the regulating transformers in accordance with the requirements of GDC 18, "Inspection and Testing of Electric Power Systems." The staff confirmed that Revision 19 to the AP1000 DCD includes the foregoing revised paragraph.

In RAI-SRP8.3.2-EEB-10, the staff remarked that Note 8 on Figure 8.3.1-4, "Inside Diesel Generator Building," (Sheets 1 and 2) of the AP1000 DCD indicates that the diesel generators include dc pre-lube oil pumps and keep-warm lube oil heaters and that these loads are not included on Table 8.3.2-2, "250 Vdc Class 1E Division B Battery Nominal Load Requirements,"

and Table 8.3.2-3, "250 Vdc Class 1E Division C Battery Nominal Load Requirements," of the DCD. Additionally, the same tables do not include dc power requirements to close and recharge the springs of the circuit breakers, nor do they include the dc power requirements for diesel generator field flashing and starting. The staff asked the applicant to indicate whether the battery sizing will include these loads or provide a reference to where these loads are powered from.

In its response dated May 7, 2009, the applicant stated that the nonsafety-related diesel generator dc loads are not powered from the safety-related batteries. The required diesel generator dc loads will be powered from the nonsafety-related dc system. Nonsafety-related breakers will also have their spring charging motors powered from the nonsafety-related dc system.

The safety-related breakers, reactor trip and RCP trip will receive their control power from the safety-related dc system. Based on the above information, the staff finds this issue resolved.

8.3.2.1.1.2 Conclusion

The staff reviewed the proposed changes to the system voltage for the operation of Class 1E dc loads from 125 Vdc to 250 Vdc and concludes that the proposed changes are acceptable.

8.3.2.3 Non-Class 1E dc and UPS System

In this section of NUREG-1793, the staff stated that the non-Class 1E dc and UPS system consists of the dc electric power supply and distribution equipment that provides dc and uninterruptible ac power to the plant non-Class 1E dc and ac loads that are needed for plant operation and investment protection. Direct current Buses 1, 2, and 3 provide 125 Vdc power to the associated inverter units that supply the ac power to the non-Class 1E UPS system. Bus 4 supplies large dc motors and other dc power loads, but not inverter loads.

In Revision 17 to the AP1000 DCD, the applicant added another dc subsystem, which includes a battery, a battery charger, and the associated dc distribution equipment, and monitoring and protection devices to serve nonsafety-related loads.

The portions of the AP1000 DCD affected include pages 8.3-18 to 8.3-20, 8.3-24, Table 8.3.2-6 (Sheet 1) and a new Table 8.3.2-6 (Sheet 2). The portions of the Tier 1 AP1000 DCD affected include Table 2.6.1-2 (untitled), Section 2.6.2, "Non-Class 1E dc and Uninterruptible Power Supply System," and Table 2.6.2-1, "Inspections, Tests, Analyses, and Acceptance Criteria," Table 2.6.2-2 (untitled), and Figure 2.6.2-1, "Non-Class 1E dc and Uninterruptable Power Supply System" (Sheet 2 of 2).

8.3.2.3.1 Evaluation

The applicant added an additional dc subsystem (EDS5) in the non-Class 1E portion of the dc and UPS system in the AP1000 design, which includes a battery, a battery charger, and the associated dc distribution equipment, and monitoring and protection devices. As a result of the addition of the non-Class 1E dc subsystem EDS5, the large dc motors that were originally powered from EDS4 will now be powered from new EDS5.

8.3.2.3.2 Conclusion

The staff has reviewed the proposed change and concludes that the addition of subsystem EDS5 provides greater flexibility to service the non-Class 1E portion of the dc and UPS system for the AP1000 design. Since the proposed change has no impact on the safety-related systems, the proposed change is acceptable.

8.4 Other Electrical Features and Requirements for Safety

8.4.1 Containment Electrical Penetrations

In the first paragraph of Section 8.4.1 of NUREG-1793, the staff stated that for modular type penetrations (three penetration modules in one nozzle), the applicant has assigned the following:

- one module for low-voltage power
- one module for 120 Vac/125 Vdc control and signal
- one module for instrumentation signal

In TR-79, the applicant deleted the above assigned module separation criteria for cables of varying voltage service levels. In addition, the applicant revised Figure 8.3.1-1 of the AP1000 DCD to correct penetration numbers associated with each RCP.

As a result of the above changes to Tier 2, the portions of the Tier 1 AP1000 DCD affected include Figure 2.6.1-1 (Sheet 1 of 4 and Sheet 3 of 4).

In addition, in Tier 1, Section 2.2.1, "Containment System," and Table 2.2.1-3, "Inspections, Tests, Analysis, and Acceptance Criteria," the applicant added a new item "6d" to address environmental qualification requirements for non-Class 1E electrical penetrations to resolve the staff's concern in RAI-TR93-ICE2-03.

8.4.1.1 Evaluation

The applicant stated that the electrical penetration conductor modules are in penetrations of the same service class. Modules for instrumentation signals will be in instrumentation penetrations; modules for power (e.g., 120/125 V) will be in control penetrations; and modules for low-voltage power will be in low-voltage power penetrations.

In addition to the above, the applicant stated that the penetration numbers shown in Figure 8.3.1-1 of Revision 15 of the AP1000 DCD were incorrect and were revised to reflect the correct penetrations associated with each RCP. The penetration numbers currently shown on the figure are E9, E10, E25, and E26. The correct penetration numbers are P10, P26, P9, and P25 for each of RCP 1B, 2B, 1A, and 2A, respectively.

With regard to qualification requirements for non-Class 1E electrical penetrations, in RAI-TR93-ICE2-03, the staff expressed its concern that non-Class 1E penetrations were not qualified for a harsh environment as were Class 1E penetrations. In its response, the applicant agreed with the staff that Class 1E and non-Class 1E penetrations must be qualified for maintaining their containment integrity to satisfy the requirements of GDC 50, "Containment Design Basis."

8.4.1.2 Conclusion

The staff has reviewed these changes and concludes that the electrical penetration conductor modules are in penetrations of the same service class. This is consistent with the recommendations of RG 1.75, and is acceptable.

The staff also concludes that including environmental qualification requirements for non-Class 1E electrical penetrations satisfies the requirements of GDC 50 and is acceptable. In addition, the staff finds the proposed change to revise the penetration numbers associated with each RCP to be administrative in nature and acceptable.

8.4.2 Reactor Coolant Pump Breakers

In the first paragraph of Section 8.4.2 of NUREG-1793, the staff stated that the RCPs are powered from the four switchgear buses located in the turbine building. Each bus powers one RCP. Variable speed drives are provided for RCP startup. Two Class 1E circuit breakers connected in series power each RCP. These are the only Class 1E circuit breakers used in the main ac power system for the specific purpose of satisfying the safety-related tripping requirements of these pumps.

In TR-79, the applicant proposed to add input and output isolation breakers to each RCP variable frequency drive (VFD) unit. The proposed change would affect AP1000 DCD Tier 2, Table 8.3.1-3 and page 8.3-53 (untitled electrical drawing).

As a result of the above changes to Tier 2, the portions of the Tier 1 AP1000 DCD affected include Figure 2.6.1-1 (Sheet 1 of 4 and Sheet 3 of 4).

8.4.2.1 Evaluation

The applicant proposed to add input and output isolation breakers to each RCP VFD unit. The applicant stated that the addition of the input and output breakers allows for the VFD unit to be completely removed from service during normal plant operation by using the bypass breaker. Without the addition of these isolation breakers, the RCP pump would need to be offline in order to service the VFD unit.

There are no regulatory requirements concerning the ability to remove VFD units during normal plant operation.

8.4.2.2 Conclusion

The staff has reviewed this change and concludes that the addition of the input and output breakers will provide the applicant flexibility to service the VFD unit without removing the RCP offline and that this change has no impact on the safety systems. Therefore, the proposed change is acceptable.

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11. ABSTRACT *(200 words or less)*

This report supplements the final safety evaluation report (FSER) for the AP10 00 standard plant design. The FSER was issued by the U.S. Nuclear Regulatory Commission (NRC) as NUREG-1793 in September 2004 , and NUREG-1793 Supplement 1 in December 2005, to document the NRC staff's technical review of the AP1000 desig n. The application for the AP1000 design was submitted on June 28, 2002, by Westinghouse Electric Corporation (Westingho use) in accordance with Subpart B, "Standard Design Certifications," of Title 10 of the Code of Federal Regulation s (10 CFR) Part 52, and Appendix 0, "Standardization of Design: Staff Review of Standard Designs." This supplement documents the NRC staff's review of Westinghouse's changes to the AP1000 design documentation in the design control document (DCD) since the issuance of Supplement 1 of the FSER. On the basis of the evaluation described in the AP100 0 FSER (NUREG-1793, NUREG-1793 Supplement 1) and this report, the NRC staff concludes that the changes to the DCD (up to and including Revision 19 to the AP1000 DCD) are acceptable and that Westinghouse's application for design certi fication meets the requirements of Subpart B to 10 CFR Part 52 that are applicable and technically relevant to the AP1000 st andard plant design.

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