6.0 ENGINEERED SAFETY FEATURES

Westinghouse has submitted information in support of its Design Certification Amendment application that Westinghouse considers "proprietary" within the meaning of the definition provided in Title 10, Code of Federal Regulations, Section 2.390(b)(5). Westinghouse has requested that this information be withheld from public disclosure and the NRC staff agrees that the submitted information sought to be withheld contains proprietary commercial information and should be withheld from public disclosure. This Chapter of the NRC staff's evaluation contains 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, containing proprietary information can be found at ADAMS Accession Number ML102410012 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) APP-GW-GLN-002 (TR-32), "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 APP-GW-GLN-106 (TR-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 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 TRs APP-GW-GLR-079, "AP1000

Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA," dated Revision 7 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 (SE) contains 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 design certification. Therefore, the proposed DCD change is acceptable pursuant to Title 10 of the Code of Federal Regulations (10 CFR) 52.63(a)(1)(vii) 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 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 CA 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 indicate 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 contained 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 by 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 by 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 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, MD, 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 for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition” (hereafter referred to as the SRP), Section 6.1.1, “Engineered Safety Features Materials,” Revision 2, issued March 2007. In addition, the staff of the U.S. Nuclear Regulatory Commission (NRC) reviewed the proposed changes as they relate to Revision 17 of the AP1000 DCD.

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 thisSEprovides 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 Westinghouse 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 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 design certification. 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 SRP Section 6.1.2. The staff concluded that the design met the quality assurance requirements of Appendix B to 10 CFR 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, 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, GDC 38, and 10 CFR 50.46(b)(5) for cooling water following a loss-of-coolant accident (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 APP-GW-GLR-079, Revision 8, "AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA."

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 100 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 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 (Agencywide Documents and Access Management System (ADAMS) Accession Number ML100950114). The staff finds it acceptable because it makes the COL information required by 6.1.3.2 consistent with the information in Section 6.1.2.1.6. **This is Confirmatory Item (CI) CI-SRP6.1.2-CIB1-01.**

Tier 1, Table 2.2.3-4, Item "x," was modified to add two requirements to the existing requirement for non-safety-related coatings used inside containment on walls, floors, ceilings, and structural steel (except for inside a chemical and volume control system 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 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 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, and pending closure of **CI-SRP6.1.2-CIB1-01**, the staff finds that the revisions proposed by Westinghouse 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 design certification. 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 degrees Celsius (C) to 30 ˚C (80 degrees 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, Westinghouse 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 DECL. There was also a negligible increase in the pressure 24 hours after the DECL 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, Westinghouse stated that DCD changes to reference a more recent WGOTHIC model, which includes the increase to external wet bulb temperature, will be included in their 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, so they are satisfied with the response. The evaluation of APP-GW-GLR-096 will be documented in Chapter 23, and will include a confirmatory item that the proposed DCD changes are acceptable.

6.2.1.1.1.3 Conclusion

Based on the evaluations by Westinghouse and the confirmatory analysis by the NRC 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, Westinghouse submitted Change Number 74 to add a vacuum relief system that replaces the operator action. The "NRC Review Package" attached to this letter contains the revised analysis and proposed DCD changes for Chapter 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 will be 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, Westinghouse 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 kPa (5 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 Westinghouse 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, Westinghouse 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 and will review the inclusion of these changes to the DCD as **CI-SRP6.2.1.2-SPCV-01**.

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. With the closure of **CI-SRP6.2.1.2-SPCV-01**, 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, Westinghouse 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. This change, tracked as **CI-SRP15.0-SRSB-02**, also adds 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 being tracked by **CI-SRP15.0-SRSB-02** 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. The staff opened **CI-SRP6.2.1.8-SPCV-01** to verify that the certified design incorporates 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, "Emergency Core Cooling," as it relates to providing abundant emergency core cooling to transfer heat from the reactor core following a LOCA

- GDC 38, "Containment Heat Removal," 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 SRP Section 6.2.2, “Containment Heat Removal Systems,” Revision 5, issued March 2007, 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 (SE), “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 SE 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 (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. Westinghouse 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. Westinghouse 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.

Westinghouse 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 #2, in the west loop compartment. The limiting HL break was the terminal end at either steam generator, due to similarity between the compartments. Westinghouse then assessed potential break locations at 5 ft intervals along each CL and HL line, as recommended by the SE 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 SE, 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 (DVI) (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 SE, 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, Westinghouse 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. Westinghouse 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, Westinghouse 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), Westinghouse considered the DEDVI line break in the loop compartment as the limiting break for unfiltered debris entry into the core.

Westinghouse 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 SE contains additional discussion of these sensitivity studies. As discussed in Section 4 of TR-26, the applicant selected the 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 SER, the hot-leg 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 Westinghouse'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 Westinghouse'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 Westinghouse'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 SE 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, reactor coolant pumps, steam generators, pressurizer, and all ASME 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 inside diameters (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 SE, 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 SE 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 SE do not address. Since no data are available for rigid cellular glass, Westinghouse used the maximum ZOI of 29 IDs from SE Table 3-2. 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 SE.

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. 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 Westinghouse jet impingement tests that were performed in 1970 by directing a [ ] 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 distance from the break to the target divided by the pipe inside diameter – was [ ] for the WCAP-7391 test conditions (cold-leg 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 ANSI/ANS 58.2-1988 jet model to calculate pressure contours for these conditions, the applicant calculated a stagnation pressure of [ ] at the concrete surface. For AP1000 hot-leg 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 hot leg and [ ] for the cold leg.

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 NRC Standard Review Plan 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 57 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. Westinghouse assumed that MRI in the ZOI degrades to pieces of crumpled foil as small as 1.3 centimeters (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. Westinghouse 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 SE. 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, (ADAMS Accession Number ML050550138), and the staff's accompanying SE (ADAMS Accession Number ML043280641)

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, "Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants," Revision 1, issued July 2000, 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 Section 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 hot-leg and cold-leg 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 SER. Briefly, the limiting CL location was the terminal end at reactor coolant pump #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 a 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 0.006 inches 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 (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 as part of design change package (DCP), DCP_NRC_002932, dated July 8, 2010. The staff is reviewing this proposed change as part of Revision 18 to the AP1000 DCD.

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 design-basis accident (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 chemical and volume control system (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 100 lb/ft^3 (1.6 grams per cubic centimeter (g/cm^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 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 0.2 feet per second (ft/s), most epoxy coating chips with a density (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 0.2 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 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 10 ft in front and 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 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 SE 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 SE discusses the test program and results.

For the reasons discussed above, the applicant's assessment using the NRC staff 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 SE. 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 (ADAMS Accession Number ML050550138), and the staff's accompanying SE (ADAMS Accession Number ML043280641).

The staff's review of Westinghouse'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 (March 31, 2008, ADAMS Accession Number ML081150379). This methodology was developed for operating pressurized-water reactors (PWRs), and the SE listed above (ADAMS Accession Number ML073521294) documents the NRC 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 (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 SE 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 (60 lbs) (27 kilograms (kg)) 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 a 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 SE 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-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 0.1 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 SE 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 NRC 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 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 Generic Letter 2004-02 evaluation. Per 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. Westinghouse 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. Westinghouse 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.

Staff review of TR-26, Table 2-1, found that while the values were consistent with the individual plant responses to Generic Letter 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 SE, 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, Westinghouse 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 SE 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 SE, 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. Per 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 SE, 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. Per the SE, 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 does 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 SE.

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. Westinghouse 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, Westinghouse 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 (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-mass per cubic foot (lbm/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). Westinghouse 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, Westinghouse 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 [ ]. Westinghouse performed the calculation for a DEDVI LOCA and for a DECL LOCA at the reactor vessel nozzle. In RAI-SRP6.2.2-SRSB-41(ADAMS Accession Number ML100330610), the staff asked that Westinghouse clarify the flow split discussion for the DECL and DEDVI breaks and related calculation presented in TR-26. Westinghouse 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 cold leg, which is greater than [ ] percent for the DEDVI break. Westinghouse assumed that (1) 90 percent of the containment water would flow through the break in the cold leg, 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. Westinghouse 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 6.6 lbs and multiplying that amount by 90 percent, yielding an estimated 5.94 lb of fiber directly entering the reactor core.

The staff also evaluated the debris transport through a hot-leg break. In RAI-SRP6.2.2-SRSB-31, the staff asked Westinghouse 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, Westinghouse developed the probable scenario that occurs during a DEHL break. During a postulated hot-leg 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 stage 4 (ADS-4) valves connected to the hot legs. 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, Westinghouse 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 hot-leg break could be captured on the upper tie plates of the core.

In its response to RAI-SRP6.2.2-SRSB-31 (ADAMS Accession Number ML100640574), Westinghouse 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 hot-leg flow conditions as tests 35, 38, and 39. Westinghouse 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, Westinghouse concluded that fibrous debris that enters through a hot-leg 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. Westinghouse 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 SE 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, Westinghouse 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 130 lbs, a total ZOI coating fine particles of 70 lbs, and a chemical debris load of 57 lbs. Of the 130 lbs of latent debris, no more than 6.6 lbs was estimated to be fibrous.

In TR-26, Westinghouse 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 Westinghouse, 90 percent of the water that enters the reactor vessel during long-term cooling comes through the cold-leg break and 10 percent comes into the reactor as filtered water through the intact DVI lines. For the purpose of these evaluations, Westinghouse 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 Westinghouse 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 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 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 NRC 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, Westinghouse 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 Westinghouse's submittal appears below.

6.2.1.8.2.7.1 LTC Sensitivity Studies to Determine Adequate Core Cooling

Westinghouse 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 SE, 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 152 lbm/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, Westinghouse 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 (ADAMS Accession Number ML100330392), the staff asked Westinghouse 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, Westinghouse 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, Westinghouse 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, Westinghouse 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 (ADAMS Accession Number ML100330610), Westinghouse 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, Westinghouse 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 hot-leg break LOCA. Because postulated debris would be introduced into the upper plenum during a hot-leg 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.

Westinghouse 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. Westinghouse used the WGOTHIC code, described in WCAP-15846, Volume 1, "WGOTHIC Application to AP600 and AP1000," Revision 1, issued March 2004, 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 Westinghouse to address these inconsistencies.

In its response, Westinghouse stated that the WGOTHIC containment analysis case was reanalyzed using corrected revised input, and the calculated sump water temperatures were within 1 ˚F of the corresponding values presented in the WCOBRA/TRAC calculation. Westinghouse 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 NRC staff reviewed the new WGOTHIC analyses and results used for Sensitivity Cases 4 through 10. Westinghouse also responded to RAI-SRP6.2.2-SRSB-25, Revision 1 (ADAMS Accession Number ML100820175), 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 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. Westinghouse 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 Westinghouse 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), Westinghouse 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 Report NP-1692, issued January 1981) to the conditions that would be simulated for Sensitivity Cases 4 through 10. In WCAP-15644-P, Westinghouse 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% 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 (ADAMS Accession Number ML101830421), Westinghouse 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 (ADAMS Accession Number ML101530051), Westinghouse 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 MVG and IFM 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 hot-leg collapsed liquid level is above the hot-leg 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 56 lbm/s versus 77.2 lbm/s in the DCD analysis. Flow into the vessel through the broken DVI line is reduced from 75 lbm/s in the DCD analysis to 55 lbm/s in Sensitivity Case 3. The core flow is reduced to 111 lbm/s with a pressure loss of 3.5 psi. The hot-leg collapsed liquid level is above the hot-leg 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 hot-leg collapsed liquid level is around the hot-leg 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 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 65 lbm/s, with a pressure loss of 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.

Westinghouse 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.

Westinghouse 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 SE section.

For Case 11, the resulting injection flow rate through the lower resistance broken DVI line (138.5 lbm/s) is greater than it is through the intact DVI line (76 lbm/s). The core flow is predicted to be 214.5 lbm/s with an average pressure loss of 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 2 psid at a flow of 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 (ADAMS Accession Number ML101470307), Westinghouse 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 No. 66 will be addressed in Chapter 23.

6.2.1.8.2.7.2 Fuel Assembly Head loss Testing

Westinghouse 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. Westinghouse 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.

Westinghouse 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.

Westinghouse had initially conducted the first 16 fuel assembly tests correlating to long-term cooling Sensitivity Cases 1 through 3. Westinghouse 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, Westinghouse increased fibrous debris loading based on the design-basis fiber amount of 6.6 lbm in the containment, equivalent to 6 lbm of fiber to the core inlet, as discussed in Section 6.2.1.8.2.6 of this SE. Westinghouse 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.

Westinghouse 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. Westinghouse prepared fibrous and particulate debris loads and chemical precipitates outside of the test loop and added them to the makeup tank per 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 subsection 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 to circulate 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 Westinghouse 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 (ADAMS Accession Number ML100640574), Westinghouse stated that most of the AP1000 fuel assembly debris head loss tests were performed with a single fuel assembly and upflow of water. Westinghouse 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, Westinghouse 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.

Westinghouse 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 cold-leg break with break flow through the downcomer and up through the core inlet region, and (2) to simulate a hot-leg 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, Westinghouse 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 Westinghouse 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.

Westinghouse 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. [

]

In the WCOBRA/TRAC analysis, Westinghouse 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. Westinghouse 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 Westinghouse discuss potential effects of non-uniform blockage in the fuel assembly test facility. In its response, dated January 29, 2010 (ADAMS Accession Number ML100330392), Westinghouse 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.

Westinghouse 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. Westinghouse 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, "Evaluation of Downstream Sump Debris Effects in Support of GSI-191," Revision 1, dated March 31, 2008. 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 cold-leg or DVI break, the limiting dP through the core is 4.1 psid with a corresponding minimum core flow rate of 65.0 lbm/s (480.7 gallon per minute (gpm) core flow rate), or [ ] per fuel assembly. Therefore, the acceptance criterion for the fuel assembly test is 4.1 psid at [ ] per fuel assembly. This is identified as acceptance criterion 1 for all fuel assembly tests not involving a hot-leg 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 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 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, Westinghouse 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 hot-leg 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 2 psid with a corresponding minimum flow rate of 214.5 lbm/s (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 hot-leg break fuel assembly tests is that the dP across the fuel assembly test article must not exceed 2 psid at a flow rate of [ ]. This acceptance criterion will be applied to each of the hot-leg break fuel assembly tests.

As discussed earlier, there are two acceptance criteria for the upflow fuel assembly tests simulating cold-leg 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 3.5 psid at fuel assembly flow rates of [ ]. Another test acceptance criterion for a hot-leg break is 2 psid at a flow rate of [ ] through the fuel assembly.

Westinghouse conducted many cold-leg 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). Westinghouse 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, Westinghouse 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 -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 (ADAMS Accession Number 100640574), Westinghouse 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, Westinghouse 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 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 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 hot leg and possibly into the upper parts of the core. In an actual hot-leg 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 cold-leg break scenario and without simulating boiling. The staff asked Westinghouse for more information and clarification concerning the hot-leg break in RAI-SRP6.2.2-SRSB-31. In response to this RAI, Westinghouse performed more testing to simulate the effects of a hot-leg break on the AP1000 core. These tests were conducted to prove that the cold-leg breaks were more limiting even if debris entering the top of the core would occur in a hot-leg break. These conditions were to be tested for an expected hot-leg break LOCA in an AP1000.

In the event of a hot-leg break, Westinghouse assumed that the post-LOCA containment debris load would be equal to 200 lbs, consisting of 6.6 lbs of fiber, 193.4 lbs of particulates, and 57 lbs of chemical precipitates. Westinghouse assumed that all the debris was transported into the core through the outer ring of fuel assemblies.

Tests 35, 38, and 39 represented the hot-leg 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 hot-leg 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 hot-leg 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, Westinghouse 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 hot-leg 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 test 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 hot-leg tests were less than the acceptance criterion of 2 psid.

In RAI-SRP6.2.2-SRSB-28 and -30, the staff asked Westinghouse 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 Westinghouse 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 response, Westinghouse prepared APP-GW-GLR-092, Revision 0 (ADAMS Accecssion Number ML100640585), 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 cold-leg break scenarios where the debris enters the core from the downcomer and lower plenum. Westinghouse 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 4.1 psid with core flow of 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 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 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 emergency core cooling system 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.

- Westinghouse 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.

- Westinghouse 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 Westinghouse 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

Westinghouse 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 (ADAMS Accession Number ML080230234).

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 SE, 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 lbm) of fiber

and 87.8 kg (193.4 lbm) 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. Westinghouse 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. Westinghouse 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, APP-GW-GLR-079, “AP1000 Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA,” 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 hot legs 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, Westinghouse 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 affect the behavior of the debris plugging the core. In partial response to RAI-SRP6.2.2-SRSB-26, Westinghouse provided TR APP-GW-GLR-110, Revision 0 (ADAMS Accession Number ML100640586), which provides an evaluation of the potential for plateout of unbuffered boric acid or buffered boric acid on the fuel rod surface. Westinghouse 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. Westinghouse 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 trisodium phosphate (TSP) that were equal to and greater than those concentrations that will occur in the AP1000 following a LOCA. Westinghouse 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 (ADAMS Accession Number ML050400463)). 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, Westinghouse 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, Westinghouse 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 LOCA 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 NRC staff accepted for GSI-191 consideration:

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

(2) The thickness of the deposition of debris, chemical precipitates, or both on the cladding (oxide + crud + precipitate) will not exceed 50 mils (1270 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 NRC staff reviewed and approved WCAP-16530-NP (ADAMS Accession Number ML073520891). 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 trisodium phosphate 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 TR 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 NRC 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 NRC 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 NRC staff finds the use of the chemical model spreadsheet contained in WCAP-16530-NP to be acceptable for determining the chemical source term for LOCADM, a Limitation and Condition was provided in the SE 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. Westinghouse 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 0.11 BTU/ (hr-ft-°F) was selected based on the possible formation of a postulated sodium aluminum silicate scale. A thermal conductivity value of 0.11 BTU/ (hr-ft-°F) is the minimum thermal conductivity value reported for sodium aluminum silicate scale. For comparison, the thermal conductivity of dry fiberglass insulation is approximately 0.05 BTU/ (hr-ft-°F), and, with eight percent of its mass wetted, it increases to approximately 0.1 BTU/ (hr-ft-°F). The NRC staff questioned if there were any materials from the thermodynamic predictions for fuel clad surface deposits which could have lower thermal conductivity values. Westinghouse responded that 0.11 BTU/ (hr-ft-°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 Westinghouse 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," January 17, 2008. (ADAMS Accession Number ML080220258)] showed thermal conductivity coefficients of representative calcium-based boiler scale deposits that were in the 0.3 to 0.5 BTU/(hr-ft-°F) range, and the thermal conductivity of glass was reported as 0.59 BTU/(hr-ft-°F).

Since the LOCADM calculations do not consider the presence of large debris, the NRC staff questioned whether small pieces of insulation ("fines") incorporated into a deposit could result in a lower thermal conductivity value than the 0.11 BTU/(hr-ft-°F) assumed for a sodium aluminum silicate scale. Westinghouse 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. Westinghouse also responded that although dry fiberglass has a lower thermal conductivity than the 0.11 BTU/ (hr-ft-°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 0.11 BTU/ (hr-ft-°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 NRC staff finds that the 0.11 BTU/(hr-ft-°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 0.11 BTU/(hr-ft-°F), the NRC staff expects the licensee to provide a technical justification for the plant-specific thermal conductivity to the NRC 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, Rev. 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 (7000 ft^3) and a large quantity of calcium silicate insulation (80 ft^3). The NRC staff questioned what additional effect the existing clad crud film and oxide scale (from three cycles) would have on the LOCADM calculations. Westinghouse 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 0.010 inches (10 mils). The maximum clad surface temperature after the start of recirculation was 324 °F, which meets the acceptance criteria of 800 °F.

Since LOCADM does not directly account for fiber fines bypassing the sump screen, the NRC staff also questioned how possible effects from fibers depositing in the core are assessed. Analysis of core inlet blockage is discussed elsewhere in this SE, 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 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 0.1 BTU/(hr-ft-°F). These calculations showed that for a chemical scale thickness of 0.050 inches (50 mils) that formed "instantaneously" at the start of recirculation, the maximum fuel clad surface temperature for a fuel rod diameter of 0.36 inches is 560 °F. Additional analyses were performed for larger diameter fuel rods, 0.416 inch and 0.422 inch OD rods. The predicted peak clad-oxide interface temperature was less than the acceptance basis value of 800 °F in each case. The NRC 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 NRC 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. Westinghouse responded that the thickness of the scale formed is limited by the amount of solids dissolved in the water. Using scale deposition models Westinghouse demonstrated that the thickest scale fragment would be insufficient to bridge a fuel rod to fuel rod span to block flow. The NRC 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 NRC 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 NRC staff finds this bump up factor to be acceptable for reasons stated in this SE section.

2. The thermal conductivity assumed for chemical scale and debris deposits represents an acceptably low value (0.11 BTU/ (hr-ft-°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 0.11 BTU/ (hr-ft-°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 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 NRC 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 NRC 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 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 0.11 BTU/(hr-ft-°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, Westinghouse 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 hot leg 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 60 lbm 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 6.6 lbm 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 (0.11 BTU/hr-ft-°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 (0.144 and 0.13 BTU/hr-ft-°F) are higher than the assumed thermal conductivity for the scale. The thermal conductivity of natural fiber, such as cotton (0.02 BTU/hr-ft-°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 (~0.40 BTU/hr-ft-°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 5.98 mils and a crud thickness of 5.51 mils, the total deposition thicknesses are [ ],

respectively. These predicted thicknesses are significantly below the acceptance criterion of 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 800 ˚F (426.7 ˚C). 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 800 ˚F (426.7 ˚C).

- The total thickness of deposition calculated for the AP1000 fuel cladding is significantly less than the acceptance value for thickness of 50 mils (1,270 μm).

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 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 lbm) total latent debris, of which up to 3.0 kg (6.6 lbm) 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.

Technical Specification 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. Technical Specification 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, Westinghouse states that it was necessary to position the containment recirculation screens next to each other due to the location of the PXS sub compartment and the large size of the screens. To address pipe rupture and jet impingement vulnerabilities, Westinghouse 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 post accident 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, contains 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 32 liters/minute (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, Westinghouse 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 SE 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 American National Standards Institute (ANSI)/American Nuclear Society (ANS)-51.1-1983, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," issued 1983, the NRC evaluated the methodology in Westinghouse topical report 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 SE 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 SE 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 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 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 NRC 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 6.8 in. ID DVI pipe and the reactor vessel DVI nozzle that has a 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 NRC'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 core makeup tank 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 automatic depressurization system 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 than 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.

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.

6.2.2 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, Westinghouse 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 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, Westinghouse 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, Westinghouse 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, and the inclusion of the proposed DCD changes will be tracked as **CI-SRP6.2.2-SPCV-02**.

- 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. The staff will track the revisions to Table 2.2.2-2 as **CI-SRP6.2.2-SPCV-20.**

In the certified DCD, Section 6.2.2.4 states that the frequency of operational testing of the PCS was consistent with the plant technical specifications and the inservice testing program. Revision 17 of the DCD removes the reference to the plant technical specifications. 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 technical specifications are used to define some of the operational test frequencies. The staff identifies withdrawal of the change as **CI-SRP6.2.2-SPCV-03**.

6.2.2.3 Conclusion

The staff finds the proposed changes to the DCD, including those being tracked as **CI-SRP6.2.2-SPCV-20** and **CI-SRP6.2.2-SPCV-03**, 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 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 containment isolation system consists of the piping, valves, and actuators that isolate the containment.

By letter dated April 5, 2006, Westinghouse 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. APP-GW-GLN-002, "AP1000 Licensing Design Change Document Zinc Addition" (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 1 in. pipe (downstream from the inside containment isolation valve) with a heavier wall 0.5 in. pipe. The staff has reviewed and approved this change (SE Section 5.2.3).

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 containment isolation system 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 SRP 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 staff will review the addition of these approved distances to DCD Table 6.2.3-1 as **CI-SRP6.2.4-SPCV-02.** 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 Revision 14 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 SRP Section 6.2.4 acceptance criterion of having a setpoint greater than or equal to 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 Westinghouse 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. The staff will confirm that the valve is shown in Tier 1, Figure 2.2.1-1 and Table 2.3.2-1, and Tier 2, Table 6.2.3-1, as **CI-SRP6.2.4-SPCV-03.**

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 Westinghouse’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 staff will review the addition of CVS-PL-V058 in DCD Tier 1, Figure 2.2.1-1 and Table 2.3.2-1, and DCD Tier 2, Table 6.2.3-1, as **CI-SRP6.2.4-SPCV-03.** Additional changes concerning distances from the containment to the outboard isolation valves will be captured in DCD Tier 2, Table 6.2.3-1 and are being tracked by **CI-SRP6.2.4-SPCV-02**.

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 Westinghouse'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), Revision 1, Westinghouse modified the elevations or locations of certain hydrogen igniters within the AP1000 hydrogen control system. Westinghouse 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 design certification 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 Westinghouse'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 Westinghouse'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 technical specifications.

6.2.6.1 Summary of Technical Information

In this SE, Tier 1, Section 2.2.1, the staff found acceptable the proposed modification to the AP1000 containment isolation system design with respect to the addition of an electrical penetration, P03.

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 AP1000 Standard Combined License TR-97, APP-GW-GLN-022, Revision 1, "DAS Platform Technology and Remote Indication Change" dated May 2007, Westinghouse 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 non safety related system. These changes necessitate the addition of a containment electrical penetration, P03. In a letter dated May 14, 2007, Westinghouse submitted responses to all the NRC RAIs on TR-97.

6.2.8.2 Evaluation

The NRC staff's assessment of the containment isolation system (CIS) design was provided in Section 6.2.4 of NUREG-1793, Final Safety Evaluation Report (FSER) for the AP1000 Design. As stated in the FSER section, the containment penetration design of isolation barriers met the following acceptance criteria of SRP 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 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 Section III, Class 2 criteria. Containment penetrations are classified as Quality Group B, as defined in RG 1.26, and seismic Category 1. The containment penetrations are identified as Class B, equivalent to ANS Safety Class 2. The staff concluded that Westinghouse 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 the SRP, including the relevant requirements of GDCs 1, 4 and 16.

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 Rev. 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, Westinghouse agreed to revise the note to state "1 of 25." The staff created confirmatory item **CI-SRP6.2.4-SPCV-04** to track this response in the next revision to the DCD.

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 report, 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 Westinghouse's proposed modification to the AP1000 containment isolation system 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, subject to the staff confirming **CI-SRP6.2.4-SPCV-04**.

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 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, technical specifications, 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 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 technical specifications 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 technical support center as the control support area. Westinghouse 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. Technical specifications 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 SE 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, Westinghouse had difficulty establishing an achievable in-leakage that could be demonstrated through testing. Originally, Westinghouse proposed a total effective in-leakage of 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, Westinghouse made a major redesign of the VES. In a letter dated May 24, 2010, Westinghouse responded to RAI-SRP6.4-SPCV-15 R1 and included all the DCD changes associated with the VES redesign. Westinghouse 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 Generic Letter 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 NRC staff approved a modification to the Standard Technical Specifications (STS) (NUREGs 1430–1434) that were proposed by the

PWR and boiling-water reactor owners groups' 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 technical specification 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 technical specifications did not contain 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 Westinghouse 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, Westinghouse 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 65 cfm for 72 hours. To meet the dose limits, the unit could only accept 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 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 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 65 cfm to an eductor to induce at least 600 cfm of air from the MCR into a filter bank to remove radionuclides. This will allow 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 15 cfm. The dose analysis assumes that 5 cfm results from ingress/egress activities, and up to 10 cfm of in-leakage can occur from all other sources. The 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 technical specifications 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 65 ± 5 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 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. The staff is tracking this change, as well as numerous other DCD changes included in the May 24, 2010 letter, in the next revision to the DCD as **CI-SRP6.4-SPCV-15.**

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 60–70 cfm would be discharged in the vicinity of the shift supervisor's officer or operator break room, and the remaining 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 (ADAMS Accession Number ML1006102070). 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 (Reference 3). 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 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 600 cfm, 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 600 cfm plus the flow from the compressed air tanks.) This allows the dose analysis to assume a constant 5-cfm in-leakage through the vestibule and up to 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 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 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 NRC staff asked the applicant to justify the surveillance frequency with a technical rationale that is based on data associated with eductor degradation.

Westinghouse 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 NRC staff finds the applicant's proposed technical specification surveillance testing frequency for the educator 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 Technical Specification 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.

Westinghouse responded in a letter date May 24, 2010 that the markup of the technical specifications that indicates the combined penetration and leakage of less than 0.5 percent is an editorial error. The applicant corrected Technical Specification 5.5.13 in the draft DCD revision pages to indicate a leakage value of less than 0.05 percent.

Per 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, Westinghouse stated that the HEPA filters will remove 99 percent of particulates consistent with the guidance in RG 1.52. The staff asked Westinghouse 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, Westinghouse 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 Technical Specification 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.

Westinghouse 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 Technical Specification 5.5.13, which shows this value to be 0.5 percent. In its response to RAI-SRP6.4-SPCV-06, Westinghouse 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.

Westinghouse 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 Technical Specification 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 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 4-in. bed. In its response to RAI-SRP-6.4-SPCV-06, Westinghouse 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 2-in.-deep charcoal bed. The applicant's proposed Technical Specification 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 NRC Generic Letter 99-02, "Laboratory Testing of Nuclear-Grade Activated Charcoal," dated June 3, 1999, (100% – Organic Iodide Efficiency)/Safety Factor = (100% – 30%)/2 = 35%). 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.

Westinghouse 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 NRC Generic Letter 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 Technical Specification 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 Technical Specification 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 technical specifications.

Westinghouse 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 FSAR Section 6.4.2.3 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, Westinghouse 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

Technical Specification 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 Technical Specification 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 Technical Specification 5.5.13.d, the staff asked Westinghouse to provide a specific citation and reference

for performing this test and to provide the required test frequency. Westinghouse responded in a letter dated June 2, 2010 to RAI-SRP6.4-SPCV-16 by stating that it would revise the proposed Technical Specification 5.5.13 to list the frequencies for each test listed in Technical Specification 5.5.13.a, b, c, and d. It removed the reference to RG 1.52, Revision 3, and ASME N510-1989 from Technical Specification 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 Technical Specification 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 (Technical Specification 5.5.13.a, b, and c). A specific RG or standard citation is not needed with the revised Technical Specification 5.5.13.d, since the test is described by the technical specification itself and the frequency is specified.

Because the frequencies given in Technical Specification 5.5.13.a, b, and c are those specified in RG 1.52 and the frequency given in Technical Specification 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. The staff is tracking this change in the next revision to the DCD as **CI-SRP6.4-SPCV-16.**

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 600 cfm, 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 600 cfm plus the flow from the compressed air tanks.) This allows the dose analysis to assume a constant 5-cfm in-leakage through the vestibule and up to 10 cfm of in-leakage from sources other than through the vestibule. The 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 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 (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.

Westinghouse 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 technical specifications sections on the MCREC and CRFA systems state, “No single active or passive failure will cause the loss of outside or re-circulated air from the CRE.”

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 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. Westinghouse 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.

Westinghouse 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 Westinghouse. In a letter dated February 25, 2010, Westinghouse submitted a response to RAI-SRP6.4-SPCV-13, Revision 2 which incorporate a bypass line for the eductor and thereby provide 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, Westinghouse 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, with the noted CIs, 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 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, Westinghouse 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, Westinghouse 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 technical support center in the certified design to the control support area. The change removed the technical support center 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 technical support center, it is reasonable for the COL applicant to want a single technical support center. Westinghouse added a statement to Tier 2* that requires the technical support center to be located in the control support area. As a result, if a COL applicant would like to locate the technical support center 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 Technical Support Center would be in the Control Support Area. Chapter 13.3 of this safety evaluation 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). 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

Technical Specification 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, Westinghouse changed the minimum decay time mandated in the technical specification to 48 hours. This technical specification 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 Technical Specification 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 technical specification 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 RTNSS systems (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.

Westinghouse 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, Westinghouse 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 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 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 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 technical specifications adequate.

6.4.3 Conclusion

The NRC staff has reviewed Westinghouse's changes to the DCD on the VES system. On the basis of the evaluation described in the AP1000 SER (NUREG-1793) and this SER, the NRC staff concludes, with the noted CIs, that the portion of the DCD on the VES system is acceptable and that Westinghouse's application to amend its design certification 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. The staff will confirm that the changes are made in the next revision of the DCD. The staff is tracking this as **CI-SRP6.4-SPCV-15** and **CI-SRP6.4-SPCV-16.** By 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 will be included in Chapter 23 of this Safety Evaluation.

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, which will be tracked as **CI-SRP6.0-SPCV-02**, 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, including those associated with CI-SRP6.0-SPCV-02, 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, Westinghouse added metallic containment (Class MC) components to the scope of the Preservice and Inservice Inspection (PSI and 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, Westinghouse revised Section 6.6.2 to delete the phrases that there are no Quality Group B and C components that require inservice inspection during operation and that relief from Section XI requirements for the baseline design certification code will not be required.

6.6.2 Evaluation

Under Revision 16 of the AP1000 DCD, Westinghouse proposed to add Class MC components to the scope of the PSI and ISI programs. 10 CFR 50.55a(b)(2)(vi) provides requirements for licensees to select their effective edition and addenda of Subsections IWE and IWL for ASME Section XI, as modified and supplemented by the requirements in paragraphs (b)(2)(viii) and (b)(2)(ix). ASME 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 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 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 Section XI, and is, therefore, acceptable.

Under Revision 17 of the AP1000 DCD, Section 6.6.2, Westinghouse proposed to remove the phrase that there are no Quality Group B and C components that require inservice inspection during operation.

Under 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 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 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.

Under Revision 17 of the AP1000 DCD, Section 6.6.2, Westinghouse 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. "

Under 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 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 Boiler and Pressure Vessel 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 under Revisions 16 and 17 meet the requirements of 10 CFR 50.55a and ASME 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.