6. ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.1 Structural Materials

The staff reviewed AP1000 Design Control Document (DCD), Tier 2, Section 6.1.1, “Metallic Materials,” for engineered safety features (ESFs), in accordance with Section 6.1.1, “Engineered Safety Features Materials,” of the Standard Review Plan (SRP). ESFs are provided in nuclear plants to mitigate the consequences of design-basis or loss-of-coolant accidents (LOCAs). General Design Criteria (GDC) 1, “Quality Standards and Record,” GDC 4, “Environmental and Dynamic Effects Design Basis,” GDC 14, “Reactor Coolant Pressure Boundary,” GDC 31, “Fracture Prevention of Reactor Coolant Pressure Boundary,” GDC 35, “Emergency Core Cooling,” and GDC 41, “Containment Atmosphere Cleanup,” of Title 10, Part 50, of the Code of Federal Regulations (10 CFR Part 50), Appendix A, “General Design Criteria for Nuclear Power Plants”; Appendix B, “Quality Assurance Criteria for Nuclear Power Plants,” to 10 CFR Part 50; and 10 CFR 50.55a apply to ESF systems.

GDC 1 and 10 CFR 50.55a(a)(1) require that structures, systems, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

GDC 4 requires that SSCs important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents (e.g., LOCAs).

GDC 14 requires that the reactor coolant pressure boundary (RCPB) shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

GDC 31 requires that the RCPB shall be designed with sufficient margin to assure that, when stressed under operation, maintenance, testing, and postulated accident conditions, it will behave in a nonbrittle manner and probability of rapidly propagating fracture is minimized.

GDC 35 requires a system to provide abundant emergency core cooling. GDC 35 also requires that during activation of the system, clad metal-water reaction be limited to negligible amounts.

GDC 41 requires that containment atmosphere cleanup systems be provided to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment. The staff’s review of the ESF structural materials was limited to ensuring that the requirements of GDC 41 were met with respect to corrosion rates as they relate to hydrogen generation to postaccident conditions.

Appendix B to 10 CFR Part 50 establishes the quality assurance requirements for the design, construction, and operation of those systems that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public.

Engineered Safety Features

This section provides a review of the materials used in the fabrication of ESF components and the need to avoid material interactions that could potentially impair the operation of the ESFs.

Summary of Technical Information

The ESFs identified in DCD Tier 2, Chapter 6 consist of the containment vessel, the passive containment cooling system (PCS), the containment isolation system (CIS), the passive core cooling system (PXS), the main control room emergency habitability system (VES), and the fission product system. DCD Tier 2, Section 6.1, describes the materials used in the fabrication of ESF components and of the provisions to avoid materials interactions that could potentially impair ESF operation for the AP1000 design.

DCD Tier 2, Table 6.1-1, "Engineered Safety Features Pressure-Retaining Materials," lists the material specifications for the principal pressure-retaining components. DCD Tier 2, Table 5.2-1, "Reactor Coolant Pressure Boundary Materials Specifications," lists the specifications for the core makeup tank (CMT), passive residual heat removal heat exchanger (PRHR HX) and valves in contact with borated water.

The materials for use in the ESF are selected for their compatibility with the reactor coolant system (RCS) and refueling water. The edition and addenda of the American Society of Mechanical Engineers (ASME) Code applied in the design and manufacture of each component are the edition and addenda established by the Design Certification. The baseline used for the DCD is the 1998 Edition, through the 2000 Addenda.

The pressure-retaining materials in ESF system components comply with the corresponding materials specifications permitted by the ASME Code, Section III, Division 1.

The components of the ESFs that are in contact with borated water are fabricated primarily from, or clad with, austenitic stainless steel or equivalent corrosion-resistant material. The use of nickel-chromium-iron (Ni-Cr-Fe) alloys in the ESFs is limited to Alloy 690/52/152.

Low- or zero-cobalt alloy hardfacing materials in contact with the reactor coolant are qualified by wear and corrosion tests for performance equivalent to Stellite-6. The use of cobalt-based alloys is minimized. Low- or zero-cobalt alloys used for hardfacing or other applications in which cobalt alloys have been previously used are qualified using wear and corrosion tests. Cobalt-free, wear-resistant alloys considered for this design include those developed and qualified in nuclear industry programs.

Austenitic stainless steel is used in the final heat-treated condition, as required by the respective ASME Code, Section II, materials specification for the particular type or grade of alloy. Austenitic stainless steel materials used in the ESF components are handled, protected, stored, and cleaned to minimize contamination that could lead to stress-corrosion cracking (SCC). These controls for ESF components are the same as those for ASME Code Class 1 components discussed in DCD Tier 2, Section 5.2.3.4. Sensitization avoidance, intergranular attack prevention, and control of cold work for ESF components are the same as the ASME Code Class 1 components discussed in DCD Tier 2, Section 5.2.3.4. Cold-worked austenitic stainless steels having a minimum specified yield strength greater than 620.5 MPa (90,000 psi) are not used for ESF components.

The material for the air storage tanks in the VES is controlled by ASME SA-372, "Specification for Carbon and Alloy Steel Forgings for Thin-Walled Pressure Vessels." It is tested for Charpy-V notch energy in accordance with supplement S3 of materials specification ASME SA-372. The material is required by the applicant's material specification to have an average of 0.51 to 0.64 mm (20 to 25 mils) of lateral expansion at the lowest anticipated service temperature. The applicant's material specification prohibits weld repairs.

The majority of the ESF insulation used in the AP1000 containment is reflective metallic insulation. Fibrous insulation may be used if it is enclosed in stainless steel cans. The selection, procurement, testing, storage, and installation of nonmetallic thermal insulation provides confidence that the leachable concentrations of chloride, fluoride, and silicate are in conformance with Regulatory Guide (RG) 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steel."

Evaluation

The staff's evaluation of the ESF materials is divided into the following four sections—(1) materials and fabrication, (2) composition and compatibility of ESF fluids, (3) component and systems cleaning, and (4) thermal insulation.

Materials and Fabrication

The staff reviewed DCD Tier 2, Section 6.1.1, for ESFs to determine the suitability of the materials for this application.

The components of the ESFs used in pressure-retaining situations are fabricated primarily from austenitic stainless steels or other corrosion-resistant material, such as Ni-Cr-Fe alloys. Where carbon steel is used in structures in contact with borated water, the steel is clad with austenitic stainless steel. Other types of protective coatings are applied to the surfaces of carbon steel structures not exposed to borated water or other fluids. Section 6.1.2 of this report reviews protective coatings. Valve seating surfaces are hardfaced to prevent failure and minimize wear.

The DCD states that the use of Ni-Cr-Fe alloy as a structural material in the ESFs will be limited to Alloy 690/52/152. The decision to use Alloy 690/52/152 was based on its improved performance in pressurized-water reactor (PWR) primary water. The staff believes the selection of Alloy 690/52/152 as the preferred nickel-based alloy is prudent because of its demonstrated improved resistance to SCC as compared to Alloy 600/82/182, which has been widely used for these applications in currently operating reactors.

Materials used in the fabrication of ESF components should be selected after consideration of the possibility of degradation during service. The materials selected for the ESF components exposed to the reactor coolant conform to Section III of the ASME Code, in particular Subarticles NB-, NC- and ND-2160, and NB-, NC- and ND-3120, as appropriate. Subarticles NB-, NC-, and ND-2160 are concerned with the deterioration of materials while in service, specifically with respect to changes in properties as distinct from loss of material. For example, valves and other components that may be made of cast austenitic stainless steel could deteriorate over time as a result of thermal embrittlement unless provisions are made to control the ferrite content. In the design of the AP1000 ESFs, the materials specifications for the pressure-retaining valves and piping in contact with the reactor coolant are the same as those used for the RCPB valves and piping. Section 5.2.3 of this report evaluates this information. In Section 5.2.3 of this report, the staff evaluated the acceptability of the RCPB materials with respect to the materials specifications, compatibility of the materials with the coolant environment, and the proposed fabrication processes. The staff concluded that the applicable NRC requirements were met.

The materials of ESF components comply with Subarticles NB-, NC-, and ND-3120 which require consideration of the effects of corrosion, erosion, and abrasive wear (Subarticles NB-, NC-, and ND-3121) and of environmental effects, specifically, irradiation-induced changes (Subarticles NB-, NC-, and ND-3124). DCD Tier 2, Section 6.1.1.2 refers to DCD Tier 2, Section 5.2.3, for discussion of the fabrication and processing of austenitic stainless steels and compliance to the regulatory positions of RG 1.31, “Control of Ferrite Content in Stainless Steel Weld Metal”; RG 1.34, “Control of Electroslag Weld Properties”; RG 1.44, “Control of the Use of Sensitized Stainless Steel”; and RG 1.50, “Control of Preheat Temperature for Welding of Low-Alloy Steel.” DCD Tier 2, Section 6.1.1.2, describes the controls placed on cold work in austenitic stainless steels by reference to DCD Tier 2, Section 5.2.3.4. The methods to control delta ferrite content in austenitic stainless steel weldments in ESF components are the same as those for the ASME Code Class 1 components described in DCD Tier 2, Section 5.2.3.4. Section 5.2.3 of this report documents the staff’s review of this section and the staff’s review of conformance with the RGs noted above. DCD Tier 2, Section 6.1.3.1, states that “the Combined License [COL] applicants referencing the AP1000 will address review of vendor fabrication and welding procedures or other quality assurance methods to judge conformance of austenitic stainless steels with RGs 1.31 and 1.44.” This is COL Action Item 6.1.1-1.

The materials selected for the AP1000 ESFs satisfy the applicable requirements of Section III of the ASME Code and Section II of the Code and therefore satisfy 10 CFR 50.55a and GDC 1. The fracture toughness of the ferritic materials will meet the requirements of the ASME Code. Cold-worked stainless steels meet the staff position that the yield strength of cold-worked stainless steels shall be less than 620.5 MPa (90,000 psi). The staff finds that the materials specifications and fabrication for the AP1000 design are acceptable and satisfy GDC 14 and 31 with regard to ensuring an extremely low probability of leakage, rapidly propagating failure, or gross rupture because they satisfy the requirements of the ASME Code and conform to the regulatory positions of the SRP and applicable U.S. Nuclear Regulatory Commission (NRC) RGs.

Composition and Compatibility of ESF Fluids

The staff reviewed DCD Tier 2, Section 6.1.1.4, “Material Compatibility with Reactor Coolant System Coolant and Engineered Safety Features Fluids,” to determine the compatibility of the ESF components with the various environments. The ESF components are manufactured primarily of stainless steel or other corrosion-resistant material. Protective coatings are applied on carbon steel structures and equipment located inside the containment. Section 6.1.2 of this report reviews protective coatings.

Austenitic stainless steel plate conforms to ASME SA-240 and is confined to those areas or components which are not subject to a postweld heat treatment. Carbon steel forgings conform to ASME SA-350. Austenitic stainless steel forgings conform to ASME SA-182. Ni-Cr-Fe alloy pipe conforms to ASME SB-167. Carbon steel castings conform to ASME SA-352. Austenitic stainless steel castings conform to ASME SA-351.

In some postulated postaccident situations, the containment could be flooded with water containing boric acid. Exposure of austenitic stainless steel to this solution for any prolonged period may induce SCC. In the design of the AP1000, the potential for this is minimized by the release of trisodium phosphate (TSP) from the pH adjustment basket into the containment sump. This action is controlled so that the pH of the sump fluid rises to above 7.0 and is thus consistent with the guidance of the NRC SRP Branch Technical Position (BTP) Materials Engineering Branch (MTEB) 6-1, “pH for Emergency Coolant Water for PWRs,” regarding protection of austenitic stainless steel from SCC. Hence, the design meets the requirements of GDC 14 for ensuring the low probability of abnormal leakage, rapidly propagating failure, or gross rupture of the RCPB boundary and safety-related structures.

DCD Tier 2, Section 6.1.1, indicated that DCD Tier 2, Section 6.2.5, contains the hydrogen production analysis for a postaccident analysis. However, this statement was incorrect because the AP1000 DCD does not contain a hydrogen generation analysis in anticipation of the NRC completion of a rule change that would eliminate the design-basis hydrogen accident. The draft safety evaluation report (DSER) stated that because this was not consistent with the current rule, the staff was not able to complete a review of the corrosion rates and consequent hydrogen generation. This was Open Item 6.1.1-1 in the Draft Safety Evaluation Report (DSER).

GDC 41 requires that containment atmosphere cleanup systems be provided, as necessary, to control fission products, hydrogen, oxygen, and other substances that may be released into reactor containment. The AP1000 design does not have a safety-related containment spray system. The staff’s review of the ESF with respect to control of hydrogen production for postaccident conditions, and thus conformance with GDC-41, was pending resolution of DSER Open Item 6.1.1-1.

The Commission approved a final rule, effective October 16, 2003, amending 10 CFR 50.44 to eliminate the requirements for hydrogen recombiners and hydrogen purge systems in currently licensed light-water reactors (LWRs). The rule relaxes the requirements for hydrogen and oxygen monitoring equipment commensurate with the equipment’s risk significance. The rule also specifies requirements for combustible gas control in future water-cooled reactors and non-water-cooled reactors.

The installation of recombiners and/or vent and purge systems previously required by 10 CFR 50.44(b)(3) was intended to address the limited quantity and rate of hydrogen generation that was postulated from a design-basis LOCA. The NRC found that this hydrogen release is not risk-significant. The rule removes the existing definition of a design-basis LOCA hydrogen release and eliminates requirements for hydrogen control systems to mitigate such a release at currently licensed nuclear power plants (NPPs) and at most future water-cooled reactors, including the AP1000. Therefore, the AP1000 is not required to perform a design basis accident hydrogen generation analysis. The applicant removed the reference in DCD Tier 2, Section 6.1.1 to a hydrogen production analysis for a post accident analysis because the hydrogen production analysis is no longer necessary. On this basis, Open Item 6.1.1-1 is resolved.

Section 6.2.5 of this report includes additional discussion related to this issue and conformance with GDC 41.

Cobalt-based alloys have limited use in the AP1000 design. In addition, cobalt-free or low-cobalt, wear-resistant alloys used in the AP1000 design are qualified by wear and corrosion tests, and include those developed and qualified in nuclear industry programs. Based on the qualification testing of these alloys and the assurance provided by performance of these or similar materials in current NPPs for this application, the staff finds the use of these alloys in the ESF design acceptable and compatible with the reactor coolant.

The materials selected for the ESFs have demonstrated satisfactory performance in operating NPPs and their selection is consistent with current practices. Corrosion is expected to be negligible on the basis of inservice observations and the results of extensive test programs. The neutron flux received by the ESF components will be sufficiently low that no irradiation-induced changes are expected. The staff finds that, given the materials selected and the chemistry controls during postaccident conditions, the ESF system should not be susceptible to SCC, and clad-metal reaction will be negligible as a result of exposure to reactor coolant and refueling water. Thus, the ESF components in the AP1000 design meet the requirements of GDC 4 and 35 and Appendix B to 10 CFR Part 50, regarding compatibility with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including LOCAs.

Component and Systems Cleaning

The staff reviewed the ESF structural materials to ensure that the requirements of Appendix B were met, as they relate to the establishment of measures to control the cleaning of material and equipment, in accordance with work and inspection instructions, to prevent damage or deterioration.

The AP1000 design conforms to RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants,” with an exception to the quality standard of the American National Standards Institute (ANSI) N45.2-1-1973 referenced in RG 1.37. The applicant referenced ASME Quality Standard NQA-2, rather than ANSI N45.2-1. The staff found this to be acceptable because the requirements of N45.2-1 have been updated and incorporated into ASME Quality Standard NAQ-2, which the staff considered an enhacement. Section 17.3 of this report discusses the staff’s evaluation of quality assurance documents. The staff finds the provisions for component and systems cleaning acceptable because these provisions conform to the regulatory positions of RG 1.37, with the exception evaluated in Section 17.3 of this report. Thus, they satisfy the quality assurance requirements of Appendix B to 10 CFR Part 50.

Thermal Insulation

The type of thermal insulation used in the AP1000 containment will be predominantly reflective metallic. Any fibrous insulation used will be enclosed in stainless steel cans. The DCD further states that any nonmetallic thermal insulation used in the design of the AP1000 ESFs will be in conformance with RG 1.36 with regard to leachable concentrations of chloride, fluoride, and silicate ions. Such actions ensure that the potential is extremely low for failure of the austenitic stainless steel pressure boundary components because of SCC resulting from the presence of contaminants in the thermal insulation. The staff finds the thermal insulation used in the AP1000 design of the ESFs to be acceptable because it conforms to the regulatory positions in RG 1.36. Thus, the provision of the type of insulation specified satisfies GDC 14 and 31 by minimizing the potential for causing SCC and thereby ensuring, with respect to this failure mechanism, that the reactor coolant boundary(RCPB) and associated auxiliary systems will have an extremely low probability of leakage, rapidly propagating failures, or gross rupture.

Conclusions

The staff concludes that the AP1000 DCD specifications concerning the materials to be used in the fabrication of the ESFs are acceptable and meet the relevant requirements of GDC 1, 4, 14, 31, 35, and 41 of Appendix A to 10 CFR Part 50; Appendix B to 10 CFR Part 50; and 10 CFR 50.55a.

6.1.2 Protective Coating Systems (Paints)—Organic Materials

Protective Coating

The staff reviewed DCD Tier 2, Section 6.1.2.1, “Protective Coatings,” in accordance with Section 6.1.2, “Protective Coating Systems (Paints)—Organic Materials,” of the SRP. The protective coating systems are acceptable if the protective coatings applied to the inside and outside of the AP1000 containment meet the requirements of Appendix B to 10 CFR Part 50, with regard to the quality assurance requirements for the design, fabrication, and construction of safety-related SSCs. To meet the requirements of Appendix B to 10 CFR Part 50, an applicant can specify that the coating systems and their applications will meet the position of RG 1.54, Revision 1, July 2000, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants.” This RG references the quality assurance standards of the American Society for Testing and Materials (ASTM) D3843-00, “Selection of Test Methods for Coatings for Use in Light-Water Nuclear Power Plants”; ASTM D3911-95, “Evaluating Coatings Used in Light-Water Nuclear Power Plants at Simulated Design-Basis Accident (DBA) Conditions”; and ASTM D5144-00, “Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants.”

Summary of Technical Information

The AP1000 design divides protective coatings into the following four areas with respect to the use of the coatings:

(1) inside containment
(2) exterior surfaces of the containment vessel
(3) radiologically controlled areas outside containment
(4) remainder of plant

In addition, the AP1000 design addresses the classification of the coatings applied inside and outside containment in DCD Tier 2, Table 6.1-2, "AP1000 Coated Surfaces, Containment Shell and Surfaces Inside Containment," based on their functions and to what extent the coatings are safety-related.

Although the DCD references RG 1.54, Revision 1, it is structured around the guidance of the RG before it was revised. The RG originally characterized coatings in terms of safety-related or non-safety-related in various spaces of an NPP. DCD Tier 2, Appendix 1A, references RG 1.54, Revision 1, and provides a summary description of the exceptions to this RG. The AP1000 design designates some coatings inside containment as non-safety-related and discusses appropriate ASTM standards that will be met. In addition, the coatings are controlled by procedures using qualified personnel, and the non-safety-related coatings are subject to the quality assurance requirements of Appendix B to 10 CFR Part 50. The AP1000 design takes exception to the RG in that the degree of conformance with the RG will be a function of the program developed by the COL applicant. DCD Tier 2, Section 6.1.3.2, states that the COL applicant will provide a program for the control of the use of these coatings, consistent with DCD Tier 2, Section 6.1.2.1.6. This is COL Action Item 6.1.2-1.

Staff Evaluation

RG 1.54, Revision 1, provides guidance on practices and programs that are acceptable to the NRC staff for the selection, application, qualification, inspection, and maintenance of protective coatings applied in NPPs. In addition, this latest revision to the RG updates the definitions of Service Level I, II, and III coatings' locations to include both safety-related and non-safety-related regions, as set forth by the ASTM Committee and the updated ASTM guidance.

By letter dated September 24, 2002, the staff, in request for additional information (RAI) 281.001, requested the applicant to address how the AP1000 design incorporates RG 1.54, Revision 1, because the terms "safety-related" and "non-safety-related" are not used in this revision to classify coatings. In addition, the staff requested the applicant to clarify which of the coatings listed in DCD Tier 2, Table 6.1-2, meet the definitions of Service Levels I, II, and III. In its response dated December 2, 2002, the applicant stated that DCD Tier 2, Section 6.1, will be revised to be consistent with the coatings classifications and associated terminology introduced in RG 1.54, Revision 1. The staff reviewed the proposed changes to DCD Tier 2, Section 6.1, including Table 6.1-2, and determined that the applicant modified this section appropriately by incorporating the new guidance in the latest revision to RG 1.54. The applicant committed to meet the guidance in RG 1.54, Revision 1, as appropriate for each of the following three service levels:

(1) Service Level I coatings are used in areas inside the reactor containment where the coating failure could adversely affect the operation of postaccident fluid systems and thereby impair safety.

(2) Service Level II coatings are used in areas where coatings failure could impair, but not prevent, normal operating performance. The functions of Service Level II coatings are to provide corrosion protection and decontaminability in those areas outside the reactor containment that are subject to radiation exposure and radionuclide contamination. Service Level II coatings are not safety-related.

(3) Service Level III coatings are used in areas outside the reactor containment where failure could adversely affect the safety function of a safety-related SSC.

The staff reviewed the COL item in DCD Tier 2, Section 6.1.3.2, and found it acceptable because the COL coatings program will conform to the NRC-accepted practice in RG 1.54, Revision 1.

Conclusions

The staff concludes that the protective coatings and their applications are acceptable and meet the requirements of Appendix B to 10 CFR Part 50. This conclusion is based on the applicant having met the quality assurance requirements of Appendix B to 10 CFR Part 50 through its commitment to RG 1.54, Revision 1. By meeting the recommendations in RG 1.54, Revision 1, the COL applicant will have evaluated the suitability of the coatings to withstand a postulated design-basis accident (DBA) environment, in accordance with NRC-accepted practices and procedures.

6.2 Containment Systems

The containment systems for the AP1000 design consist of the following three components:

(1) a steel vessel as the primary containment

(2) a shield building surrounding the primary containment which provides external missile protection and is also a principal component of the PCS

(3) supporting systems

The primary containment prevents the uncontrolled release of radioactivity to the environment and acts as the passive safety-grade interface to the ultimate heat sink.

The primary containment has a design leakage rate of 0.10 weight percent per day of the original containment air mass per day following a DBA. This value is determined by the containment design pressure of 508.12 kPa (59 psig). The limiting calculated peak pressure occurs for a double-ended, guillotine break in the RCS cold-leg LOCA, and is 499.84 kPa (57.8 psig).

As the interface to the ultimate heat sink (the surrounding atmosphere and external cooling water), the primary containment is an integral component of the PCS described in Section 6.2.2 of this report. The exterior of the containment vessel provides a surface for evaporative film cooling and works in conjunction with the natural draft airflow created by the shield building baffle and chimney arrangement to reduce the pressure and temperature of the containment atmosphere following a DBA.

6.2.1 Primary Containment Functional Design

The AP1000 primary containment consists of a 39.62-m- (130-ft-) diameter cylindrical steel shell with ellipsoidal upper and lower heads and a nominal wall thickness of 4.45 cm (1.75 in.). The wall thickness is increased to 4.76 cm (1.875 in.) in the transition region where the cylindrical shell enters the concrete embedment to provide a margin against corrosion. The wall thickness is also increased near primary containment penetrations to structurally compensate for these openings. The primary containment will enclose the nuclear steam supply system (i.e., reactor vessel, steam generators (SGs), reactor coolant pumps (RCPs), pressurizer, and associated connecting piping), the in-containment refueling water storage tank (IRWST), the CMTs, the accumulator tanks, and the refueling canal. Additionally, the primary containment houses associated mechanical support components; electrical support components; and heating, ventilation, and air conditioning (HVAC) support components.

The primary containment shell is supported by embedding the lower head between the concrete of the containment internal structures and the concrete encasement external to the containment vessel. No structural connection exists between the free-standing portion of the containment and the adjacent structures, other than penetrations and their supports, and the supports for the baffle wall of the PCS. Thus, the portion of the cylindrical primary containment shell above the support region elevation of 30.48 m (100 ft) is structurally independent.

The primary containment has a net free volume of 58,333 m^3 (2,060,000 ft^3) and is designed to withstand pressures and temperatures resulting from a spectrum of primary coolant and steamline pipe breaks. The primary containment design parameters consist of an internal design pressure of 508.12 kPa (59 psig) and a design temperature of 149 °C (300 °F).

The following AP1000 containment design features are compared to those of the AP600 design in Table 6.2-1 of this report:

- containment structure type
- power level
- containment free volume
- design pressures
- design temperatures
- calculated peak DBA containment pressures and temperatures
- heat removal systems

- hydrogen control systems
- containment penetrations

Table 6.2-1 Comparison of AP600/AP1000 Containment Design Features

Parameter	AP600	AP1000
Power, Megawatt thermal (MWt)	1940	3400
Type of containment structure	4.1-cm- (1.625-in.-) thick steel cylindrical primary containment with top and bottom dome, surrounded by concrete shield building	4.45-cm- (1.75-in.-) thick steel cylindrical primary containment with top and bottom dome, surrounded by concrete shield building
Secondary containment	No	No
Free volume	4.9 E+4 m^3 (1.73E+6 ft^3)	5.8 E+4 m^3 (2.06E+6 ft^3)
Volume-to-power ratio	25.2 m^3/MW (892 ft^3/MW)	17.2 m^3/MW (606 ft^3/MW)
Internal design pressure	411.6 kPa (45 psig)	508.12 kPa (59 psig)
External design pressure	20.7 kPa (3.0 psid)	20 kPa (2.9 psid)
Design temperature	138 °C (280 °F)	149 °C (300 °F)
Design leak rate, weight %/day	0.10	0.10
Calculated peak internal pressure (design margin)	405.4 kPa (44.1 psig) (1.5%)	499.84 kPa (57.8 psig) (1.6%)
Calculated peak external pressure (design margin)	13.8 kPa (2.0 psid) (33%)	16.6 kPa (2.4 psid) (17%)
Heat removal system	PCS and non-safety-grade fan coolers	PCS and non-safety-grade fan coolers
Combustible gas control system	(1) DBA—passive autocatalytic recombiners (2) severe accidents—hydrogen igniters	(1) Defense-in-depth—passive autocatalytic recombiners (2) severe accidents—hydrogen igniters
Number of penetrations	~40	~40

Parameter	AP600	AP1000
Motive power for containment isolation valves	(1) air-operated valves (2) Class 1E dc motor-operated valves	(1) air-operated valves (2) Class 1E dc motor-operated valves

Section 15.3 of this report describes the staff's evaluation of the ability of the AP1000 design to comply with the relevant dose limits of 10 CFR 50.34 and GDC 19, "Control Room." That evaluation assumed a 0.10 weight percent per day leak rate from the AP1000 containment. Operating plants have demonstrated the ability to verify a design leak rate as low as 0.10 weight percent per day. Therefore, the staff finds that a design leak rate of 0.10 weight percent per day is acceptable for the AP1000.

The AP1000 design does not have safety-related containment sprays, which makes natural deposition on surfaces in containment far more important than in past designs. The design does include non-safety-related containment sprays, as described in DCD Tier 2, Section 6.5.2, and evaluated by the staff in Section 19.2.3.3.9 of this report.

Section 6.2.1.1 of this report discusses the containment design pressure margin. The design capability of the AP1000 for external pressure is 20 kPa (2.9 psid). Westinghouse calculated a peak external pressure of 16.6 kPa (2.4 psid).

The reliance of the AP1000 on cooling by naturally occurring physical phenomena represents a significant difference from designs for currently operating reactors. The heat removal system for the AP1000 containment is the PCS, which is described in detail in Section 6.2.2 of this report. A principal feature of the system is that it relies on gravity-driven flow and natural circulation to perform its cooling function. Previously licensed Westinghouse plants use containment sprays and fan coolers, which rely on active components (i.e., pumps and fans) to function.

The AP1000 has nonsafety, passive autocatalytic recombiners, as described in DCD Tier 2, Section 6.2.4, to provide for defense-in-depth protection against the buildup of hydrogen following a LOCA. Section 6.2.5 of this report includes the staff's evaluation of the combustible gas control.

Table 6.2-1 of this report also shows that the AP1000 containment has considerably fewer mechanical penetrations (approximately 40) than a typical two-loop design (approximately 100). Additionally, the containment isolation valves in the AP1000 are primarily either air operated or motor operated from a safety-grade direct current (dc) power source. In previous designs, containment isolation valves were typically motor-operated valves (MOVs) powered from safety-grade alternating current (ac). Section 6.2.4 of this report contains the staff's evaluation of the CIS.

Compliance with Regulatory Requirements

The Westinghouse AP1000 containment evaluation model is based on assumptions that maximize the initial stored energy within containment and minimize the rate of heat transfer from containment. The approach is consistent with the guidance provided in SRP Section 6.1.1.2.A, “PWR Dry Containments, Including Subatmospheric Containments.” Westinghouse uses the WGOTHIC 4.2 computer program to evaluate the containment performance. Chapter 21 of this report provides a review of WGOTHIC 4.2 and the model used to evaluate containment performance.

Compliance with Appendix A to 10 CFR Part 50

Section 6.2 of the SRP delineates the current guidance for demonstrating that a containment design complies with GDC 16, “Containment Design,” GDC 38, “Containment Heat Removal,” and GDC 50, “Containment Design Basis.” The SRP addresses acceptance criteria and some specific model assumptions for design-basis LOCA and main steamline break (MSLB) analyses for all existing containment types. Westinghouse elected to evaluate the PCS performance using these current guidelines. The Westinghouse documentation for the AP1000 evaluation model is consistent with the guidelines in SRP Sections 6.2.1 and 6.2.1.1.A, as well as RG 1.70, “Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.” Westinghouse also uses approved methods for the LOCA and MSLB mass and energy releases, and follows the guidance provided in SRP Sections 6.2.1.3 and 6.2.1.4, respectively.

Peak Pressure Criteria (GDC 16 and 50)

Acceptance criteria for existing containments include a margin between the design pressure and a conservatively calculated peak accident pressure. NUREG-0800, Section 6.2.1.1.A states that the margin should vary from 10 percent at the construction permit (CP) stage to a peak calculated pressure “less than the containment design pressure” at the operating license (OL) stage. Thus, even in instances in which much data and information are known and the staff possesses an independent, confirmatory calculational capability, a 10-percent margin was expected at the CP stage to cover uncertainties in meeting the requirements of GDC 16 and 50 following final construction (i.e., at the OL stage).

For the AP1000 containment, Westinghouse proposed a criterion that the calculated peak accident pressure not exceed the design pressure (a zero-margin criterion). In meeting this criterion, Westinghouse stated that it uses a conservative approach consistent with current staff guidelines. For design certification, under 10 CFR Part 52, the staff does not necessarily need the same demonstration of margin as normally expected at the CP stage. An appropriate initial test program, combined with appropriate inspections, tests, analyses, and acceptance criteria (ITAAC), is in place to assure that the assumptions and performance characteristics of the AP1000 containment and the PCS, as used in the licensing analyses, are verified prior to operation. Therefore, the staff finds the applicant’s approach to be acceptable.

The staff reviewed the differences between the AP600 and the AP1000 to assure that the WGOTHIC 4.2 computer program and evaluation model are applicable to the AP1000. This review included the modeling assumptions, the treatment of stratification and circulation, and the applicability of the AP600 phenomena identification and ranking table (PIRT), scaling and testing program, and the applicability of the mass and heat transfer correlations for the larger AP1000. Chapter 21 of this report documents the staff's review.

The staff has determined that the WGOTHIC 4.2 computer program, combined with the conservatively biased evaluation model described in WCAP-15846, Revision 1, "WGOTHIC Application to AP600 and AP1000," is acceptable for the evaluation of the peak containment pressure following a DBA for the AP1000 design, as discussed in Chapter 21.6 of this report. Although the WGOTHIC 4.2 code itself is essentially a best-estimate tool, Westinghouse has taken a conservative approach in the evaluation methodology it is using to support design certification. The AP1000 WGOTHIC evaluation model uses appropriately conservative input values and applies conservative multipliers on the correlations used for PCS heat and mass transfer. Conservative models are used in the AP1000 WGOTHIC evaluation model to address the following areas:

- lumped-parameter network representation
- noncondensible circulation and stratification
- PCS flow and heat transfer models
- dead-ended and liquid-filled compartments

During the peak pressure period (up to about 2400 seconds for a LOCA, and up to about 1000 seconds for an MSLB), these conservatisms compensate for the uncertainties introduced by the use of passive safety features, leading to an overall conservative result for the calculated peak containment pressure.

Long-Term Pressure Analysis (GDC 38)

The objective of the long-term pressure analysis is to demonstrate that the containment design conforms to the requirements of GDC 38, which requires containment temperature and pressure be maintained at acceptably low levels following any LOCA.

In Item II.b of Section 6.2.1.1.A of the SRP, the staff guidance used to evaluate compliance with GDC 38, is that the containment pressure should be reduced to less than 50 percent of its peak value within 24 hours of the occurrence of a design-basis LOCA. This assures that the containment leak rate used for the siting evaluation is consistent with the design basis analysis assumption. Westinghouse proposed that the calculated pressure reduction be based on 50 percent of the design pressure to be consistent with current guidelines related to GDC 38. The staff found this approach acceptable because the peak calculated pressures are near the design value.

The Westinghouse analytical procedure can credit the effect of two-dimensional (2-D) heat conduction (between wet and dry regions of the containment shell) when less than full PCS water coverage of the containment shell is expected. The procedure was first presented in May 1997 (Westinghouse letter NSD-NRC-97-5152, "AP600 Design Changes to Address Post 72-Hour Actions," Attachment 2 - Description of method to account for circumferential (2-dimensional) conduction through the steel containment shell for containment pressure analyses, dated May 23, 1997), and discussed at an Advisory Committee on Reactor

Safeguards (ACRS) meeting in December 1997 (Westinghouse letter NSD-NRC-97-5492, "Presentation Material for December 9, 10, 11 and 12, 1997 ACRS, Meeting," dated December 17, 1997). Westinghouse did not identify, or at least account for, the need to consider 2-D heat transfer for the long-term containment pressure response when the PCS flow rate decreases after the passive containment cooling water storage tank (PCCWST) water level drops below 6.19 m (20.3 ft) in the selection of the analysis methodology (GOTHIC) and in the development of a model for the PCS (WGOTHIC). With the coverage area less than the initial assumed 90 percent, heat transfer from the hot, dry regions of the shell into the cooler, wet regions of the shell would occur. To account for this modeling deficiency, Westinghouse performed an ancillary calculation to credit more PCS water in the evaporation process, effectively generating a correction factor, and applied it to the limited PCS flow model.

The staff evaluated this ancillary calculation, and the staff believes that there is a real effect from 2-D heat conduction. However, as an insufficient amount of test data is available to validate this model, the staff is unable to determine how much credit should be given in evaluating the AP1000 design performance after 24 hours when 2-D heat conduction is included in the analysis. Therefore, the results from a WGOTHIC analysis that includes the 2-D heat conduction enhancement to the evaporation limited flow model may not be used to demonstrate reduced containment leakage after 24 hours when performing dose assessments, as described in DCD Tier 2, Section 15.6.5.3.3.

The 2-D enhancement to the Evaporation Limited flow model may not be used to credit leakage reduction for siting evaluations. A separate analysis may be performed for the limiting LOCA without 2-D conduction, and included in DCD Tier 2 Section 6.2.1.1.3, "Design Evaluation." This separate analysis may be used to confirm the assumption used in DCD Tier 2, Section 15.6.5.3.3 of reducing the containment leakage to half its design value after 24 hours.

After the peak pressure period, the uncertainty in the treatment of heat transfer processes continues to increase. These uncertainties, resulting from the evaluation model treatment of non-condensable circulation and stratification and the effectiveness of the PCS cooling at a reduced flow rate, are difficult to quantify using the available test data. Nevertheless, the heat removal capability of the AP1000 PCS (as calculated by the WGOTHIC Evaluation model) is sufficiently greater than the decay power to conclude that the containment pressure will decrease. The WGOTHIC analyses demonstrate the effectiveness of the PCS to reduce the containment pressure and maintain that pressure below the design limit. The system safety function to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels has been demonstrated.

The applicant credits containment pressure reduction when performing dose assessments for siting evaluations. Therefore, a separate analysis was performed for the limiting LOCA without 2-D conduction, as discussed in DCD Tier 2, Section 6.2.1.1.3. This separate analysis confirmed the assumption used in DCD Tier 2, Section 15.6.5.3.3 of reducing the containment leakage to half its design value after 24 hours.

Compliance with 10 CFR 52.47(b)(2)

The unique characteristics of the PCS are explicitly recognized in the regulations governing the evaluation of standard plant designs. As stated in 10 CFR 52.47(b)(2)(i)(A), in the absence of a prototype plant that has been tested over an appropriate range of normal, transient, and accident conditions, the following requirements must be met for a plant that "utilizes simplified, inherent, passive, or other innovative means to accomplish its safety functions":

(1) The performance of each safety feature of the design has been demonstrated through either analysis, appropriate test programs, experience, or a combination thereof.

(2) Interdependent effects among the safety features of the design have been found acceptable by analysis, appropriate test programs, experience, or a combination thereof.

(3) Sufficient data exist on the safety features of the design to assess the analytical tools used for safety analyses over a sufficient range of normal operating conditions, transient conditions, and specified accident sequences, including equilibrium core conditions.

Consistent with these requirements, the passive plant vendor, Westinghouse, developed and performed design certification tests of sufficient scope, including both separate-effects and integral-systems experiments, to provide data with which to assess the computer programs used to analyze plant behavior over the range of conditions described in item 3 above.

To satisfy the requirements of 10 CFR 52.47(b)(2)(i)(A), Westinghouse developed test programs to investigate the passive containment safety systems. These programs included both component and phenomenological (separate-effects) tests and integral-systems tests. The cold water distribution test was a full-scale representation of the PCS flow characteristics. Additional separate-effects tests have been performed to extend the range of existing mass and heat transfer correlations used in the analysis codes, to comply with the last of the three requirements above.

The large-scale test (LST) is the only integral test for the PCS. Because this test facility exhibited a number of shortcomings in scaling and prototypicality, the LST data was not used in an integral mode. Instead, the LST data was used in a separate-effects mode to demonstrate the conservatism of portions of the evaluation model. The staff concludes that sufficient data has been provided to establish that the evaluation model is conservative at the scale of the AP1000.

The staff agrees with the Westinghouse PIRT conclusions that the difference between the AP600 and the AP1000 does not change the ranking of the phenomena, that no new phenomena have been identified, and that the models developed to address the high- and medium-ranked phenomena for the AP600 remain applicable for the AP1000 (see Chapter 21 of this report). The staff also agrees with the Westinghouse conclusion that the mass and heat transfer correlations are acceptable for the evaluation of the AP1000, and that the AP600 test program adequately covers the expected ranges for which these correlations are used (see Chapter 21 of this report).

The staff concludes that the evaluation model contains sufficient conservatisms, including factors to compensate for shortcomings in the LST, to accept WGOTHIC, in combination with the evaluation model for DBA licensing analyses to support design certification, as discussed in Chapter 21 of this report. Section 21.6.5.8.3 of this report defines the calculational method that has been reviewed by the staff and found acceptable with respect to the SRP 6.2.1 Section IV, "Evaluation Findings," item 1d finding. For any future licensing analyses, the AP1000 nodal model described in Section 13 of WCAP-15846, Revision 1, "WGOTHIC Application to AP600 and AP1000," should be used. Further, the assumptions should be consistent with the limitations and restrictions denoted in Section 21.6.5.8.3 of this report. As discussed above, the 2-D enhancement to the Evaporation Limited flow model may not be used to credit leakage reduction for siting evaluations. A separate analysis may be performed for the limiting LOCA without 2-D conduction, and included in DCD Tier 2, Section 6.2.1.1.3. This separate analysis may be used to confirm the assumption used in DCD Tier 2, Section 15.6.5.3.3 of reducing the containment leakage to half its design value after 24 hours.

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

The staff reviewed the temperature and pressure response of the primary containment to a spectrum of LOCAs and MSLBs, and completed a review of the minimum containment backpressure for LOCA analyses. Westinghouse did not analyze the response of the shield building because this structure is vented to the atmosphere and is not designed to maintain a set pressure under LOCA or MSLB conditions.

The Containment Analytical Model

Westinghouse calculated the short- and long-term pressure and temperature response of the containment using the WGOTHIC computer code in the lumped-parameter mode. WGOTHIC is a program for modeling multiphase flow. It solves the conservation equations, in integral form, for mass, energy, and momentum for multicomponent flow. The momentum conservation equations are written separately for each phase in the flow field (drops, liquid pools, and atmosphere vapor). The following terms are included in the momentum equation:

- storage
- convection
- surface stress
- body force
- boundary source
- phase interface source
- equipment source

In creating the WGOTHIC 4.2 computer program, used for licensing analyses to support the design certification, from the GOTHIC computer program, Westinghouse added analytical models to represent the unique features of the AP1000 containment. Major additions included modeling the condensation heat transfer in the presence of noncondensible gases on the interior wall of the containment, one-dimensional heat conduction through the containment wall, and heat rejection on the exterior of the containment shell via evaporative cooling, natural convection cooling, and radiative cooling.

Design features of the PCS to address post-72 hour actions, in response to the staff requirements memorandum (SRM) dated January 15, 1997, relating to SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design," have been incorporated into the AP1000 design. These include an on-grade PCS auxiliary water storage tank and two recirculation pumps that provide the required makeup flow to the PCCWST from the auxiliary tank for the post-72 hour period (up to 7 days). In addition, the PCCWST also provides makeup to the spent fuel pool and for fire protection.

Table 6.2-2 of this report provides the initial conditions of pressure, temperature, humidity, and net containment free volume used for the DBA analyses.

Table 6.2-2 Containment Initial Condition

Parameter	**Initial value**	
Internal Temperature	48.9 °C	120 °F
Pressure	108.2 kPa	15.7 psia
Relative Humidity	0 %	0 %
Net Free Volume	58,333 m^3	2.06E+6 ft^3
External Temperature	46.1 °C dry bulb 26.7 °C wet bulb	115 °F dry bulb 80 °F wet bulb

The current initial internal pressure and temperature, and the external temperature are technical specifications (TS) maximums and have been shown, in Section 5 of WCAP-15846, Revision 1, to result in a conservative peak pressure calculation. It was also shown that zero percent relative humidity is a conservative assumption. The staff has reviewed these input assumptions and finds them acceptable because they maximize the calculated peak containment pressure, consistent with the guidance in SRP Section 6.2.1.1.A.

Table 6.2-3 of this report provides the PCS flow rates and surface area coverage used for the DBA safety analyses (from DCD Tier 2, Table 6.2.2-1).

Table 6.2-3 PCS Flow Rates and Area Coverage

PCCWST Water Elevation		**Safety Analysis Flow Rate**		**Area Coverage**
ft	m	Liters/min	gpm	% of circumference
27.5	8.38	1775.7	469.1	90
24.1	7.35	857.8	226.6	90
20.3	6.19	667.4	176.3	72.9
16.8	5.12	545.9	144.2	59.6
(Note)		381.2	100.7	41.6

Note: from passive containment ancillary water storage tank, at 72 hours.

WGOTHIC models the passive heat sinks in the containment, one-dimensional heat transfer through the containment vessel, evaporation of cooling water from the exterior of the containment vessel, and radiative and natural convection heat transfer in the shield building annulus. 2-D conduction can be considered in WGOTHIC 4.2 analysis to account for heat transfer between wet and dry regions of the containment shell for the long-term pressure response, when the PCS water coverage fraction is reduced as a result of lower PCS water delivery rates, as shown in Table 6.2-3 of this report. The passive heat sinks include both concrete and steel structures inside the containment, which can absorb energy from the containment atmosphere. The energy source is modeled using information from a table of mass and energy releases included in DCD Tier 2, Sections 6.2.1.3 and 6.2.1.4.

Containment Pressure Response

The staff has reviewed the Westinghouse analyses of the pressure response of the AP1000 containment, as discussed below.

Internal Pressure Analysis

The pressure response of the AP1000 containment can be divided into two temporal phases—the short-term or blowdown portion of the transient, and the longer term representing the remainder of the transient. The AP1000 containment response to the high-pressure blowdown portion of LOCA and MSLB transients is not significantly different from that of a standard Westinghouse two- or three-loop plant. Blowdown is the time during which the contents of the coolant system are expelled through a postulated break. During blowdown, the large time constant for heat transfer through the containment shell causes the AP1000 containment response to be governed primarily by the energy absorbed by pressurizing the internal containment volume and by heat removal via internal structures (heat sinks). Therefore, the predicted containment response during the blowdown phase should be similar to that for a standard Westinghouse two- or three-loop plant. None of the new AP1000 passive design features come into play during this first portion of a postulated transient. In Section 8 of WCAP-15846, Revision 1, Westinghouse performed an analysis during the blowdown portion of the LOCA to compare the current multinode model to a simple, single-node model (similar to the modeling used for currently operating reactors). This analysis showed that the multinode model during blowdown yields results comparable to the simple, single-node model.

The long-term portion of the transient begins after the coolant system has blown down. During this time, the mass and energy releases are greatly reduced, and the PCS begins operating and transferring energy stored inside the containment to the ultimate heat sink. The primary mechanism of heat removal from inside the containment is the condensation of steam on the inside of the containment shell. This heat is ultimately rejected to the environment by means of radiative, convective, and evaporative cooling from the containment outer surface.

For the LOCA events, two limiting, double-ended guillotine RCS pipe breaks are analyzed. In one case, the break is postulated to occur in the hot-leg of the RCS; in the other case, the break is in the cold-leg. The hot-leg break results in the highest blowdown peak temperature. The cold-leg break results in the highest postblowdown peak pressure. The cold-leg break analysis includes the long-term contribution to containment pressure from the sources of stored energy, such as the SGs. Section 6.2.1.3 of this report discusses the LOCA mass and energy release calculations.

For the MSLB event, a representative pipe break spectrum is analyzed. The WGOTHIC 4.2 code is used to analyze various break sizes, power levels, and failure assumptions. Section 6.2.1.4 of this report discusses the MSLB mass and energy release calculations.

Table 6.2-4 of this report provides a summary of the calculated pressures and temperatures for LOCA and MSLB postulated accidents.

Table 6.2-4 Summary of Calculated Pressures and Temperatures for a LOCA and an MSLB Using WGOTHIC 4.2

Break	**Peak Pressure [kPa (psig)]**	**Available Margin[1] [kPa (psig)]**	**Peak Temperature[2] [°C (°F)]**	**Pressure at 24 hours [kPa (psig)]**
LOCA, double-ended, hot-leg guillotine	446.06 (50.0)	62.05 (9.0) 12.2%	213.6 (416.5)	- - -
LOCA, double-ended, cold-leg guillotine	499.84 (57.8)	8.27 (1.2) 1.6 %	140.5 (284.9)	253 (22)[3]
MSLB, 0.13 m^2 (1.4 ft^2), full double-ended rupture (DER), 101% power, MSIV failure	471.57 (53.7)	36.5 (5.3) 7.2%	190.7 (375.3)	- - -
MSLB, 0.13 m^2 (1.4 ft^2), full DER, 30% power, MSIV failure	496.39 (57.3)	11.7 (1.7) 2.3%	189.9 (373.9)	- - -

Notes:
1. design pressure is 508.12 kPa (59 psig), margin determined by absolute pressure
2. localized temperature in the break compartment (node)
3. Value includes 2-D multiplier. A separate analysis was performed without the 2-D multiplier to confirm the pressure at 24 hours was less than half the design value. Without the 2-D multiplier the pressure was calculated to be 26.5 psig.

The maximum calculated pressure in the primary containment occurs from a double-ended, guillotine break in the RCS cold-leg LOCA, and is 499.84 kPa (57.8 psig) at about 1800 seconds after the LOCA begins. This value provides a margin of 1.6 percent to the design pressure of 508.12 kPa (59 psig).

The WGOTHIC 4.2 containment evaluation model was created using assumptions that maximize the initial stored energy within containment and minimize the rate of heat transfer from containment. A summary of the conservatisms Westinghouse identified by the AP1000 WGOTHIC 4.2 containment evaluation model is as follows:

- The mass and heat transfer coefficients on the inner containment vessel surface are multiplied by a factor of 0.73. Only free convection is considered on the inner surface. The multiplier was determined by an assessment of the LST and separate tests, as discussed in Section 21.3.4 of this report.

- The mass and heat transfer coefficients on the outer containment vessel surface are multiplied by a factor of 0.84. Mixed convection is considered on the outer surface. The multiplier was determined by an assessment of the LST and separate tests, as discussed in Section 21.3.4 of this report.

- The vessel wall emissivity values are reduced by 10 percent to reduce the radiation heat transfer.

- The maximum outside air temperature of 46 °C (115 °F) is used as a boundary condition to reduce the heat transfer from containment and is consistent with the TS maximum allowable ambient temperature.

- The maximum containment air temperature of 49 °C (120 °F) and internal pressure of 108.2 kPa (1 psig) are used as initial conditions and are consistent with the TS limits. An initial condition of zero percent humidity is used to increase the initial stored energy inside containment.

- A single failure of one out of three valves controlling the PCS cooling waterflow is assumed. This assumption provided the minimum PCS liquid filmflow rate.

- The PCS liquid filmflow is credited only following a 337-second delay. This corresponds to the time needed to establish a steady liquid film coverage pattern based on the AP600 initial flow rate of 1666 Liters/min (about 440 gpm). The higher initial flow rate for the AP1000, 1775.7 Liters/min (about 469.1 gpm), helps to offset the increased height of the AP1000 containment wall and would result in a shorter delay time. However, Westinghouse has maintained the 337-second delay for the AP1000 licensing analyses, as described in Section 7 of WCAP-15846, Revision 1.

- The water coverage is obtained from the limiting flow model, as described in Section 7 of WCAP-15846, Revision 1, based on the wetted surface areas listed in Table 6.2-3 of this report. 2-D conduction can be considered in the limiting flow model to account for heat conduction from the dry to wet regions of the containment shell when the PCS water coverage is reduced if the water level in the PCCWST falls below 6.19 m (20.3 ft).

- A 0.051-cm (20-mil) air gap is assumed between the steel liner and the concrete on applicable internal heat sinks.

- The loss coefficient in the external annulus includes a 30-percent increase over the value derived from the test program.

- Condensation and convection on heat sinks in the dead-ended compartments, below the operating deck, are not credited after the blowdown period (about 30 seconds after accident initiation). This conservative assumption is also employed for MSLB analyses.

- Heat transfer to horizontal, upward-facing surfaces that may become covered with a condensation film is not credited.

Table 6.2-4 of this report summarizes the limiting LOCA and MSLB peak containment pressure and temperature calculations provided in support of the design certification.

The staff has allowed Westinghouse to move the heat sink information from DCD Tier 2 (previously provided in Table 6.2.1.1-4, "Metal Heat Sinks"; Table 6.2.1.1-5, "Concrete Heat Sinks"; Table 6.2.1.1-6, "Containment Shell and Baffle Heat Sinks"; and Table 6.2.1.1-7, "Shield Building Concrete Heat Sinks") by reference to Section 13 of WCAP-15846, Revision 1, which is considered to be fully proprietary to Westinghouse Electric Company.

Summary of Staff CONTAIN Analyses

The staff performed independent confirmatory analysis with the CONTAIN 2.0 computer code for the limiting LOCA and limiting MSLB cases (Memorandum from J. Rosenthal, RES, to J. Hannon, NRR, "AP1000 Containment DBA Calculations Using the CONTAIN Code," dated March 26, 2003). These analyses indicate similar characteristics for the PCS performance in both the limiting LOCA and limiting MSLB events. The calculated peak pressure for the MSLB was about 3.4 kPa (0.5 psi) higher than the WGOTHIC value. For the LOCA case, the calculated peak pressure was about 2.76 kPa (0.4 psi) lower than the WGOTHIC value. The acceptability of the PCS is based on the results of the WGOTHIC analyses.

Long-Term Internal Pressure Analysis

The objective of the long-term internal pressure analysis is to demonstrate that the design is consistent with the design requirements of GDC 38, which provide for containment temperature and pressure to be maintained at acceptably low levels following any LOCA.

In Item II.b of Section 6.2.1.1.A of the SRP, the staff guidance used to evaluate compliance with GDC 38, is that the containment pressure should be reduced to less than 50 percent of its peak value within 24 hours of the occurrence of a design-basis LOCA. This assures that the containment leak rate used for the siting evaluation is consistent with the design basis analysis assumption. Westinghouse proposed that the calculated pressure reduction be based on 50 percent of the design pressure to be consistent with current guidelines related to GDC 38. The staff found this approach acceptable since the peak calculated pressures are near the design value.

Westinghouse presented the results of an analysis for the long-term containment pressure resulting from the design-basis LOCA, including the 2-D correction, to demonstrate the desired result, that the long-term (post 24-hour) pressure remains below 50 percent of the design pressure. (See DCD Tier 2, Figure 6.2.1.1-7.) This analysis is for the cold-leg break LOCA.

This LOCA case also results in the most limiting peak containment pressure of 499.84 kPa (57.8 psig) at approximately 1,800 seconds into the event.

DCD Tier 2, Figure 6.2.1.1-9, provides the response for the limiting hot-leg break LOCA. For the MSLB, the pipe break spectrum analysis has identified the full double-ended rupture at 30 percent power as the limiting break with respect to peak containment pressure. DCD Tier 2, Figure 6.2.1.1-1 illustrates this response. This limiting MSLB case yields a peak containment pressure of 496.39 kPa (57.3 psig) at approximately 810 seconds into the event. The containment pressure rises until the secondary side blowdown is complete. Once blowdown is completed, there is no additional mass or energy released to containment. With no mass and energy source, the containment pressure rapidly decreases as the internal heat sinks and PCS continue to absorb energy. The 2-D heat conduction enhancement to the evaporation limited flow model is not applicable to the MSLB because the accident is terminated following blowdown prior to the time the enhancement would be applied, after the first PCCWST standpipe uncovers.

DCD Tier 2, Table 6.2.1.1-3 provides the calculated pressure for the most limiting DBA. This table demonstrates that the long-term containment pressure following the limiting LOCA is maintained at an acceptably low level as required by GDC 38.

The staff considers the AP1000 PCS design to be in compliance with GDC 38. The system safety function to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptably low levels has been demonstrated. The staff believes that there is a real effect from 2-D heat conduction. However, as an insufficient amount of test data is available to validate this model, the staff is unable to determine how much credit should be given in evaluating the Westinghouse design performance after 24 hours. A separate analysis was performed for the limiting LOCA without 2-D conduction, as discussed in DCD Tier 2, Section 6.2.1.1.3, “Design Evaluation.” This separate analysis confirmed the assumption used in DCD Tier 2, Section 15.6.5.3.3 of reducing the containment leakage to half its design value after 24 hours. Although the containment pressure response is different from current licensed plants, the PCS is acceptable and consistent with the passive design objectives on which the AP1000 PCS is based.

External Pressure Analysis

The staff reviewed the analysis conducted to determine the maximum external pressure, or reverse differential pressure, that would result from design-basis events or inadvertent system actuations. Conformance with the criteria of SRP Section 6.2.1.1.A, “Containment Functional Design—PWR Dry Containments, Including Subatmospheric Containments,” forms the basis for concluding whether the Westinghouse maximum external pressure analysis satisfies the following requirement:

- GDC 16, as it relates to the reactor containment and associated systems being provided to assure that containment design conditions important to safety are not exceeded for as long as postulated accident conditions require

Engineered Safety Features

The worst case scenario presented by Westinghouse for the maximum external pressure was the loss of all ac power sources during extreme cold weather. The pressure evaluation was conducted using the WGOTHIC code, and assumed that all ac power sources were lost, resulting in a reduction of heat generated in containment. An ambient temperature of −40 °C (−40 °F) and a steady 21.5 m/sec (48 mph) wind outside of containment were also assumed, thereby maximizing the cooling of the containment atmosphere and maximizing the differential pressure across the containment vessel. Other analytical assumptions were as follows:

- An initial internal containment temperature of 49 °C (120 °F) was assumed, to maximize the heat transfer from the containment wall and maximize the pressure differential across the containment vessel.

- An initial internal relative humidity of 100 percent was assumed, to minimize the air in containment, allowing for a greater reduction in pressure from the condensation of steam.

- An initial containment pressure of 99.97 kPa (or −0.2 psig) was assumed, consistent with the TS limiting condition for operation (LCO).

- No air leakage into the containment was assumed during the transient.

The calculated differential pressure across the containment vessel is approximately 16.6 kPa (2.4 psid). The design external pressure is 20 kPa (2.9 psid). To mitigate the event, Westinghouse states in DCD Tier 2, Section 6.2.1.1.4, that containment pressure instruments (four total) would indicate the containment pressure, and operators could open the containment ventilation purge isolation valves, which are powered by Class 1E batteries, to restore containment pressure. Westinghouse further states in the DCD that operators would have sufficient time to restore the pressure before reaching the design external pressure limit.

The staff notes that because the AP1000 has no safety-related ac power, the loss of all ac power is not a beyond-design-basis event, as it would be for a plant with safety-related ac. Events involving inadvertent PCS actuation, failed fan cooler controls, malfunction of containment purge valves, drainage of the IRWST into containment, prolonged operation of the ejector in the primary sample system, and the maximum ambient temperature change were considered, but Westinghouse found them nonbounding.

In plants with active containment engineered safety heat removal systems, such as sprays, the inadvertent actuation of these systems would cool the containment, reduce the pressure, and result in the bounding event. The inadvertent actuation of the PCS with the containment fan coolers in operation is not considered to be a bounding event. The chilled water supply and return lines to the containment recirculation cooling system fan coolers isolate following any event resulting in a containment isolation signal to provide containment integrity. Operation of the containment fan coolers is limited by the minimum temperature, 4.4 °C (40 °F), of the chilled water system. The maximum heat transfer from containment for the external pressure transient was chosen without PCS operation because the heated water within the PCS water storage tank (minimum temperature of 4.4 °C (40 °F)) would tend to heat the containment shell, particularly at the elevated flow rates for the first few hours when compared to the extreme cold temperature, −40 °C (−40 °F). The staff finds that Westinghouse has identified the most limiting case with regard to the maximum reverse differential pressure.

In an SRM dated June 30, 1997, the Commission approved the staff's recommendation that the AP600 include a containment spray system, or equivalent, for accident management following a severe accident. The AP1000 design also includes a containment spray system for accident management following a severe accident. DCD Tier 2, Section 6.5.2, describes the containment spray system, and Section 19.2.3.3.9 of this report includes the staff's evaluation of the system.

As noted in DCD Tier 2, Section 6.5.2.1.4, the use of the containment spray during power operation requires multiple failures of closed valves, including a locked closed valve outside of containment, and a remotely operated valve inside containment, from the MCR or remote access workstation. Therefore, the staff finds inadvertent actuation of the containment spray system during power operations not credible. Inadvertent spray actuation does not need to be considered for the external pressure evaluation.

During shutdown modes, the containment isolation valves are open and the header for the fire protection water inside containment is pressurized. When the header inside containment is pressurized, an additional manual valve between the header and the remotely operated valve on the line to the spray ring is closed. During shutdown modes, the pressure in the fire protection header is caused by the head of water in the PCS storage tank on the roof of the shield building. Pressurization of the spray ring by the water storage tank would result in flow through the nozzles, but insufficient flow to produce a spray. To produce spray from the spray ring, a fire pump must be operating and the appropriate valves open to the containment fire protection header. The connection from the fire pumps to the containment header is normally closed with a manual valve located outside containment. Therefore, the staff finds inadvertent actuation of the containment spray system during shutdown operations not credible, and inadvertent spray actuation does not need to be considered for the external pressure evaluation.

On the basis of its review, the staff finds that Westinghouse has identified the bounding event (loss of all ac power sources during extreme cold weather) for the maximum external containment pressure. Westinghouse has satisfied GDC 16 by providing acceptable margin between the maximum calculated reverse differential pressure and the design differential pressure, and has stated that operators would be able to restore containment pressure before the reverse differential pressure design limit is reached. This provides assurance that containment design conditions important to safety are not exceeded for the duration of accident conditions. The staff, therefore, finds the Westinghouse maximum external pressure analysis acceptable.

6.2.1.2 Subcompartment Analysis

The staff reviewed the analysis conducted to determine the maximum differential pressure, or loading, that the containment subcompartment walls would be subjected to as a result of the most limiting postulated line break within a particular subcompartment. Conformance with the criteria of SRP Section 6.2.1.2, "Subcompartment Analysis," and SRP Section 6.2.1.3, "Mass and Energy Release Analysis For Postulated Loss-of-Coolant Accidents," forms the basis for concluding whether the Westinghouse subcompartment analysis satisfies the following requirements:

- GDC 4, regarding the appropriate protection of SSCs important to safety against dynamic effects that may result from equipment failures

- GDC 50, regarding the ability of the reactor containment structure and its internal compartments to accommodate the calculated pressure and temperature conditions resulting from any LOCA

<u>Selection of Postulated Breaks and Subcompartments</u>

As discussed in DCD Tier 2, Section 6.2.1.2, Westinghouse applied the leak-before-break (LBB) concept to the RCS high-energy piping. LBB is applicable to RCS piping 15.24 cm (6 in) in diameter or greater. The general concept of LBB is that piping for which LBB has been demonstrated to be applicable, by deterministic and experimental methods, would leak at a detectable rate from postulated flaws before catastrophic failure of the pipe would occur as a result of loads experienced under normal, anticipated transient, and safe-shutdown earthquake (SSE) conditions. Application of LBB to the containment subcompartment analysis allows the postulated rupture of large pipes to be precluded from the spectrum of postulated breaks. DCD Tier 2, Section 3.6, "Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping," summarizes the LBB evaluation.

GDC 4 states, in part, that "dynamic effects associated with postulated pipe ruptures...may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping." Therefore, for the LBB concept to be acceptable with respect to subcompartment analysis, the applicant must demonstrate that the probability of a particular rupture is extremely low under design-basis conditions. Section 3.6.3 of this report discusses the staff's evaluation and acceptance of LBB for the AP1000.

Table 6.2-5 of this report summarizes the postulated breaks and design pressures for the subcompartments analyzed. For all subcompartments, the postulated breaks envelop other line breaks that could be postulated to rupture (in accordance with the size limits of LBB) in the particular area.

Table 6.2-5 Postulated Breaks and Subcompartment Design Pressures

Subcompartment	Postulated Break	Design Pressure
Steam generator compartment and access area	10.16-cm (4-in.) pressurizer spray line 10.16-cm (4-in.) SG blowdown line 7.62-cm (3-in.) RCS cold-leg pipe 7.62-cm (3-in.) RCS hot-leg pipe	34.5 kPa (5 psid)

Subcompartment	Postulated Break	Design Pressure
Pressurizer valve room	10.16-cm (4-in.) pressurizer spray line	34.5 kPa (5 psid)
CVS room	7.62-cm (3-in.) RCS cold-leg pipe	34.5 kPa (5 psid)
CVS pipe tunnel	10.16-cm (4-in.) SG blowdown line	51.7 kPa (7.5 psid)
Maintenance floor and operating compartment walls	0.093-m^2 (1-ft^2) main steamline rupture	34.5 kPa (5 psid)

Westinghouse performed an evaluation of rooms which could have either a main or startup feedwater line break. No significant pressurization of the rooms is expected to occur because the postulated breaks are located in regions which are open to the large free volume of the containment. For these regions, the main or startup feedwater line breaks are not limiting.

The reactor vessel cavity was analyzed for asymmetric pressurization resulting from a 18.9 Liters/min (5-gpm) leak rate crack in the primary piping. The reactor vessel cavity was not analyzed for asymmetric loading from pipe breaks because all of the piping in the reactor vessel cavity is qualified to LBB, which also applies to the weld joining the RCS piping in the vessel cavity and the “safe-ends,” or nozzles, attached to the reactor vessel. The staff's acceptance of LBB in Section 3.6.3 of this report encompasses pipe welds and breaks at weld locations that do not need to be postulated for LBB piping for the purpose of the subcompartment pressurization analysis.

The pressurization loads for the IRWST are determined by the pressure and hydrodynamic loads from the discharge of the first, second, and third stage of the automatic depressurization system (ADS), as discussed in DCD Tier 2, Section 3.6.1, “Postulated Piping Failures in Fluid Systems Inside and Outside Containment.” Westinghouse conducted an analysis to determine the hydrodynamic loading on the IRWST due to ADS discharge. Section 6.2.8 of this report discusses the staff’s review and acceptability of this analysis.

Differential Pressure Analysis

To obtain the fluid mass and energy released from the postulated breaks, Westinghouse used the modified Zaloudek correlation to calculate the critical mass flux for the 7.62-cm (3-in.) cold-leg break, the 7.62-cm (3-in) hot-leg break, and the 10.16-cm (4-in.) SG blowdown line break. For the 10.16-cm (4-in.) pressurizer spray line break, the Fauske breakflow model in NOTRUMP was used. The modified Zaloudek correlation used for pipes other than the pressurizer spray line helps create a smooth transition between subcooled and saturated flow regimes when the pressure in the break element exceeds the saturation pressure. With the modified Zaloudek correlation, Westinghouse assumed the mass flux to remain constant at initial full-power conditions to maximize the mass and energy release, resulting in a conservatively large release to the containment.

NOTRUMP is used for certain breaks. NOTRUMP better models the more complex depressurization that occurs with the vapor and subcooled liquid that is released through both sides of the pressurizer spray line break. The NOTRUMP piping model does not include friction losses, which result in a higher pressure at the break and thus a greater mass release, which is conservative. Because NOTRUMP conservatively models the AP1000 depressurization, the staff finds this acceptable.

Westinghouse chose the initial conditions of the subcompartment atmosphere to maximize the calculated differential pressures. These include use of the maximum allowable air temperature, minimum pressure, and minimum relative humidity.

Westinghouse used the TMD computer code, described in WCAP-8077, “Ice Condenser Containment Pressure Transient Analysis Methods”; WCAP-8077, March 1973 (proprietary); and WCAP-8078 (nonproprietary) to calculate the differential pressure across the subcompartment walls. It assumed 100-percent entrainment of fluid droplets because this yielded the largest differential pressure. Westinghouse used the unaugmented critical flow model option in TMD to predict the critical mass flow rate between nodes. Furthermore, no credit was taken for vent paths which become available only after the break occurs, such as blowout panels, doors, and collapsing insulation.

The staff finds that the TMD modeling assumptions meet the guidance in SRP Section 6.2.1.2. In particular, this guidance provides for the following:

- The nodalization should be chosen so that substantial pressure gradients do not exist within a node, and 100-percent entrainment should be assumed.

- Vent flow should be based on homogeneous mixture in thermal equilibrium with 100-percent water entrainment.

- The maximum allowable air temperature, minimum pressure, and minimum relative humidity should be assumed for initial conditions.

Some of the subcompartments may not meet the 40-percent pressure margin specified in SRP Section 6.2.1.2, for the CP stage of a review. However, the calculations show margins still exist. At the OL stage of a review, the SRP guidance is that the peak differential pressure should not exceed the design pressure. The staff has determined that the few exceptions to the 40-percent margin are acceptable for design certification.

The staff reviewed the short-term mass and energy release data, and the methodology as it applies to the AP1000. The staff finds that Westinghouse meets the guidance provided in SRP Section 6.2.1.3 regarding the mass and energy release used in the analysis by assuming a constant mass blowdown rate and using an acceptable choked flow correlation. With regard to the choked flow model, the staff has previously found use of the modified Zaloudek coefficient acceptable through its approval of WCAP-8264, “Westinghouse Mass and Energy Release Data for Containment Design”; WCAP-8264-P-A, June 1975 (proprietary); and WCAP-8312-A, Revision 2, August 1975 (nonproprietary). Furthermore, SATAN-VI has been found acceptable through the staff's review of WCAP-10325, “Westinghouse LOCA Mass and Energy Release

Model for Containment Design—March 1979 Version" and WCAP-10325, May 1983 (proprietary). NOTRUMP has been found acceptable for use in currently licensed plants for small line breaks, as discussed in Chapter 21 of this report.

Although the staff approved the TMD and SATAN-VI codes used for subcompartment analysis for previously licensed plants, it reviewed the use of these codes as they apply to the AP1000, as well as the modeling assumptions made by Westinghouse.

In SRP Section 6.2.1.1.B, "Ice Condenser Containments," the staff found the TMD code acceptable for subcompartment analyses, provided that the unaugmented critical flow model was used. While the AP1000 is not an ice condenser containment, the staff has previously found TMD acceptable for non-ice condenser operating plants.

The staff finds the correlations, computer codes, and methodologies used by Westinghouse acceptable for the AP1000 subcompartment pressurization analysis.

In conclusion, the staff finds that Westinghouse has satisfied GDC 4 with regard to containment subcompartments by considering the dynamic effects of postulated pipe ruptures within subcompartments. Consistent with GDC 4, Westinghouse has shown, by analysis, that pipe breaks above a certain size can be precluded from that piping for which breaks must be postulated. Furthermore, Westinghouse has satisfied GDC 50 by designing containment subcompartment walls to withstand, with appropriate margin, the calculated differential pressures resulting from pipe breaks postulated in accordance GDC 4. Therefore, the staff finds the Westinghouse containment subcompartment pressurization analysis acceptable.

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

Westinghouse documented mass and energy releases for two different types of transients, the subcompartment differential pressure analysis and the containment integrity analysis. The first analysis (mass and energy release analyses in support of the subcompartment differential pressure analysis) was referred to as a short-term analysis because it focused on blowdown. The staff evaluated these releases and found them acceptable with the criteria of SRP Section 6.2.2, "Subcompartment Analysis," and SRP Section 6.2.1.3, "Mass and Energy Release Analysis For Postulated Loss of Coolant Accidents" (see Section 6.2.1.2 of this report).

The second type of analysis described the methodology used to determine the releases for the containment pressure and temperature calculations using the WGOTHIC code (referred to as the long-term analysis). These releases were used for the containment integrity analysis discussed in Section 6.2.1.1 of this report.

The long-term analysis considered the limiting break size for containment integrity analysis and the LOCA design basis as the complete, double-ended guillotine severance of the largest RCS pipe. The release rates were calculated for pipe failure at two locations (the hot-leg and the cold-leg). These break locations were analyzed for both the short-term and the long-term transients. Because the initial operating pressure of the RCS is approximately 15,513 kPa (2,250 psi), the mass and energy would be released extremely rapidly when a break occurs. As the water exits from the broken pipe, a portion of it would flash to steam because of the differences in pressure and temperature between the RCS and containment. The RCS would depressurize rapidly because breakflow would exit on both sides of the pipe.

Long-Term Mass and Energy Release Data

A long-term LOCA analysis calculational model is typically divided into the following four phases:

(1) blowdown, which includes the period from the accident initiation (when the reactor is in a steady-state, full-power operation condition) to the time that the broken loop pressure equalizes to the containment pressure

(2) refill, which is the time from the end of the blowdown to the time when the ECCS refills the vessel lower plenum

(3) reflood, which begins when the water starts to flood the core and continues until the core is completely quenched

(4) post-reflood, which is the period after the core has been quenched and energy is released to the RCS primary system by the RCS metal, core decay heat, and the SGs

The Westinghouse long-term analysis considered only the blowdown, reflood, and post-reflood phases of the transient. The refill period is omitted from the analyses because Westinghouse assumed that the refill period occurred immediately upon the end of blowdown, so that the releases to the containment were maximized. This assumption is consistent with the guidance provided in SRP 6.2.1.3, Section II.3.c.

The AP1000 long-term LOCA mass and energy releases were predicted for the blowdown phase for postulated double-ended cold-leg and double-ended hot-leg breaks. The blowdown phase mass and energy releases were calculated using the SATAN-VI computer code (see WCAP-10325-P-A (proprietary) and WCAP-1032S-A (nonproprietary), May 1983).

The staff reviewed the long-term LOCA mass and energy release data, and the methodology as it applies to the AP1000. This methodology is described in Section 14, “LOCA Mass and Energy Release Calculation Methodology,” of WCAP-15846, Revision 1. The staff has determined that the SATAN-VI LOCA blowdown computer program is acceptable for use in obtaining LOCA mass and energy releases for the LOCA blowdown phase for containment analyses. SATAN-VI has been approved by the staff for this purpose, as discussed in SRP Section 6.2.1.4, and models the AP1000 passive safety features in a conservative manner. The postblowdown mass and energy releases back into the containment atmosphere from the accumulators, CMTs, and IRWST injection into the RCS were found to be acceptable. The increased mass and energy released from the primary system is consistent with the guidance in SRP Sections 6.2.1.4 and 6.2.1.1.A to maximize the calculated containment pressure and temperature. In the AP1000, for LOCA analyses, the break location switches to the fourth-stage ADS at about 1500 seconds into the limiting LOCA scenario.

Energy Sources

The following energy sources were accounted for by Westinghouse in the long-term LOCA mass and energy calculation:

- decay heat
- core stored energy
- RCS fluid and metal energy
- SG fluid and metal energy
- accumulators
- CMTs
- IRWST
- zirconium-water reaction

Westinghouse employed the following assumptions to analyze the core energy release for maximum containment pressure:

- maximum expected operating temperature

- allowance in initial temperature to account for instrument error and deadband

- margin in RCS volume (+1.4 percent)

- allowance in volume for thermal expansion (+1.6 percent)

- 100-percent, full-power operation

- allowance for calorimetric error (+1.0 percent of full power)

- conservatively modified coefficients of heat transfer, which ensure that RCS metal and SG stored energies are released at a conservatively high rate

- allowance in core stored energy for fuel densification effect

- margin in core stored energy (+15.0 percent)

- allowance in initial pressure to account for instrument error and deadband

- margin in SG mass inventory (+10.0 percent)

- 1 percent of the zirconium around the fuel reacts

The staff reviewed the methods and assumptions used to release the various energy sources during the blowdown phase. The staff found the methods and assumptions increase the stored energy in the primary system consistent with the guidance in SRP Sections 6.2.1.4 and 6.2.1.1.A to maximize the calculated containment pressure and temperature. Therefore, the methods and assumptions are acceptable for the licensing analyses.

Engineered Safety Features

Description of Blowdown Model

Westinghouse employed the SATAN-VI model to determine the mass and energy released from the RCS during the blowdown phase of a postulated LOCA. The model is described in WCAP-10325, dated May 1983.

Description of Postblowdown Model

Westinghouse used the mass and energy inventories at the end of blowdown to define the initial conditions for the beginning of the reflood portion of the transient. The broken and unbroken loop SG inventories were kept separate to account for potential differences in the cooldown rate between the loops. In addition, the mass added to the RCS from the IRWST was returned to containment as breakflow so that no net change in system mass occurred.

Energy addition from decay heat was computed using the 1979 American Nuclear Society (ANS) standard (plus 2 sigma) decay heat table. The energy release rates from the RCS metal and SG metal were modeled using exponential decay rates, which generally exhibit an initial rapid energy release followed by a significantly slower, gradual release of energy.

The accumulator, CMT, and IRWST mass flow rates are computed from the end of blowdown to the time the tanks empty. The rate of RCS mass accumulation is assumed to decrease exponentially during the reflood phase. More CMT and accumulator flow is spilled from the break as the system refills. The breakflow rate is determined by subtracting the RCS mass addition rate from the sum of the accumulator, CMT, and IRWST flow rates.

The primary differences between the AP1000 design and currently operating Westinghouse PWRs are the ESFs. The safety features of currently operating plants include passive and active systems, while the AP1000 safety features are only passive. However, this difference only affects long-term inventory makeup systems and not the system behavior during the blowdown phase. The only safety feature which participates during blowdown is the accumulator system, which is included in both current plants and the AP1000 and is modeled with the NRC-approved LOCA mass and energy release methodology. The AP1000 uses spherical accumulators, whereas currently operating Westinghouse-designed plants use cylindrical accumulators. The accumulator inventory is depleted well before the time of peak pressure, so any difference in discharge rate associated with the different accumulator geometry would have an insignificant effect on the calculation for peak containment pressure. The gravity-driven CMTs do not operate in the blowdown timeframe and are not included in the SATAN-VI model. CMTs cannot inject into the common direct vessel injection (DVI) line against the pressure of the gas-charged accumulators during the blowdown phase of the accident. Therefore, the methodology for calculating the mass and energy release to containment during the blowdown is not affected by the AP1000 passive systems.

The variable noding structure of the SATAN model allows the user to simulate current and advanced RCS geometries with generalized control volumes. The standard Westinghouse PWR RCS noding was modified to specifically model the AP1000 RCS geometry. This modeling included two cold-legs in the broken loop and the DVI line to the downcomer.

No changes in the approved, conservative design-basis methodology or modeling assumptions, as described in WCAP-10325-P-A, have been made to the SATAN-VI code to model the AP1000. The behavior of the release of the initial RCS inventory during the initial blowdown for the AP1000 is identical to currently operating plants. The flexibility of the noding structure in a SATAN-VI model allows for an accurate representation of the AP1000 geometry.

Therefore, the SATAN-VI code is acceptable for predicting the mass and energy releases during the blowdown phase for the AP1000 design.

Mass that is added to and remains in the vessel is assumed to be raised to saturation. Therefore, the actual amount of energy available for release to the containment for a given time period is determined from the difference between the energy required to raise the temperature of the incoming flow to saturation and the sum of the decay heat, core stored energy, RCS metal energy, and SG mass and metal energy release rates. The energy release rate for the available breakflow is determined from a comparison of the total energy available release rate and the energy release rate, assuming that the breakflow was 100 percent saturated steam. Saturated steam releases maximize the calculated containment pressurization.

The staff reviewed the postblowdown model as it applies to the AP1000. The staff found the postblowdown model increases the mass and energy released from the primary system consistent with the guidance in SRP Sections 6.2.1.4 and 6.2.1.1.A. Therefore, the model is acceptable for the licensing analyses because it maximizes the calculated containment pressure and temperature.

Single-Failure Analysis

The assumptions for the containment mass and energy release analysis are intended to maximize the calculated release. For the LOCA mass and energy releases, a single failure could reduce the flow rate of water to the RCS, but would not disable the passive core cooling function. For example, if one of the two parallel valves from the CMT were to fail to open, the injection flow rate would be reduced and, as a result, the break mass release rate would decrease. Therefore, to maximize the releases, the AP1000 mass and energy release calculations conservatively do not assume a single failure. The effects of a single failure in the PCS are taken into account in the containment analysis.

Containment Response Analysis and Initial Conditions

Westinghouse employed the WGOTHIC computer code to determine the containment response following a LOCA. Section 6.2.1.1 of this report discusses the staff's review of the initial conditions for LOCA analyses, the WGOTHIC code, and its results.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Rupture Inside Containment

A steamline rupture occurring in containment releases significant amounts of high-energy steam to the containment environment, resulting in high containment temperatures and pressures which may challenge design limits. Various break sizes and power levels are analyzed to determine the limiting break case for containment integrity. Steamline breaks are postulated to occur with the plant in any operating condition ranging from hot shutdown to full power. Because SG mass decreases with increasing power level, breaks occurring at a lower power generally result in a greater total mass release to the containment. Because of increased energy storage in the primary system, increased heat transfer in the SGs, and additional energy generation in the nuclear fuel, the energy released to the containment from breaks postulated to occur during power operation may be greater than for breaks occurring with the plant in a hot-shutdown condition. Additionally, steam pressure and the dynamic conditions in the SGs change with increasing power. This has significant influence on both the rate of blowdown and the amount of moisture entrained in the fluid leaving the break following an event.

Break area is also important when evaluating steamline breaks. It controls the rate of releases to the containment, and influences the steam pressure decay and the amount of entrained water in the blowdown flow. The MSLB analysis used to determine the limiting break case for peak containment pressure was found to be a full, double-ended pipe rupture downstream of the steamline flow restrictor. For this case, the actual break area equals the cross-sectional area of the steamline, but the blowdown from the SG with the broken line is controlled by the flow restrictor throat area (0.13 m^2 (1.4 ft^2) nominal). The reverse flow from the intact SG is controlled by the smaller of the pipe cross-section, the steam stop valve seat area, or the total flow restrictor throat area in the intact SG. The reverse flow has been conservatively assumed to be controlled by the flow restrictor in the intact loop SG.

Because of the opposing effects of changing power level on steamline break releases, no single power level can be identified as a worst case initial condition for a steamline break event. Therefore, several different power levels spanning the operating range, as well as the hot shutdown condition were analyzed, including 101-percent, 70-percent, 30-percent, and 0-percent power.

The effects of the assumption of the availability of offsite power are enveloped in the analysis. Offsite power is assumed to be available where it maximizes the mass and energy released from the break because of the following:

- The continued operation of the RCPs, until automatically tripped as a result of CMT actuation, maximizes the energy transferred from the RCS to the SG.

- The continued operation of the feedwater pumps and actuation of the startup feedwater system, until they are automatically terminated, maximizes the SG inventories available for release.

The AP1000 is equipped with a passive safeguards system, including the CMT and the PRHR HX. Following a steamline rupture, these passive systems are actuated when their setpoints are reached. This decreases the primary coolant temperatures. The actuation and operation of these passive safeguards systems do not require the availability of offsite power.

When the PRHR is in operation, the core-generated heat is dissipated to the IRWST by means of the PRHR HX. This causes a reduction of the heat transfer from the primary system to the

SG secondary system, resulting in a reduction of mass and energy releases because of the break.

The availability of ac power, in conjunction with the passive safeguards system (CMT and PRHR), maximizes the mass and energy releases resulting from the break. Therefore, blowdown occurring in conjunction with the availability of offsite power is more severe than for cases in which offsite power is not available.

Analyses that considered single active failure of either one main steamline isolation valve (MSIV) or one feedwater isolation valve determined that the main feedwater isolation valve failure was not limiting. The spectrum of cases analyzed to determine the limiting MSLB event all assume the failure of one MSIV.

The containment response to the MSLB event is determined by the magnitude and duration of the mass and energy releases, the containment volume, steam/air circulation to the heat sinks, and time response of the heat sinks. Because of the nature of the secondary-side releases discussed in the previous section, the MSLB transient is characterized by the addition of superheated steam to the containment throughout the transient. Consistent with the guidance established in NUREG-0588, “Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment,” a value of 8-percent revaporization is assumed for all MSLB transients analyzed.

The containment pressure continues to rise until the secondary-side blowdown is complete. Once blowdown is complete, no additional mass or energy is released to the containment. With no mass and energy source, the containment pressure decreases rapidly as the internal heat sinks and PCS continue to absorb energy.

The pipe break spectrum analysis has identified the full double-ended rupture at 30-percent power as the limiting break with respect to peak containment pressure. This limiting MSLB case yields a peak containment pressure of 496.39 kPa (57.3 psig) at about 810 seconds into the event.

<u>Significant Parameters Affecting Steamline Break Mass and Energy Releases</u>

The following four major factors influence the release of mass and energy following a steamline break:

(1) SG fluid inventory
(2) primary-to-secondary heat transfer
(3) protective system operation
(4) the state of the secondary fluid blowdown

The following is a list of plant variables that have a significant influence on the mass and energy releases:

- plant power level
- main feedwater system design

- startup feedwater system design
- postulated break type, size, and location
- availability of offsite power
- safety system failures
- SG reverse heat transfer and RCS metal heat capacity

The staff reviewed the significant parameters affecting steamline break mass and energy releases as they apply to the AP1000 and found them acceptable because they maximize the calculated peak containment pressure, consistent with the guidance in SRP Section 6.2.1.1.A.

Description of Blowdown Model and Mass and Energy Release Data

In the AP1000 analysis, Westinghouse employed the blowdown models described in WCAP-8822, "Mass and Energy Releases Following a Steamline Rupture," by R.E. Land, dated September 1976. The LOFTRAN-AP computer program is used to determine the mass and energy releases from steamline breaks ("LOFTRAN and LOFTTR2 AP1000 Code Applicability Document," WCAP-14234, Revision 1 (proprietary), June 1997).

The above-cited methodologies reflect current technology by including the effect of SG superheat. The staff reviewed the application of these methodologies to the AP1000 and found them to be acceptable because they are consistent with the guidance provided in the SRP Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Rupture," Section II, to produce a conservative result.

Containment Response Analysis and Initial Conditions

Westinghouse employed the WGOTHIC computer code to determine the containment response following a steamline break. Section 6.2.1.1 of this report discusses the staff's review of the initial conditions for steamline break analysis, the WGOTHIC code, and its results.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies of the ECCS

The staff reviewed the analysis conducted to determine the minimum containment pressure that could exist during the period of time until the core is reflooded following a LOCA. It conducted this review to confirm the validity of the pressure used as a boundary condition in the ECCS performance studies. Conformance with the criteria of SRP Section 6.2.1.5, "Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies," forms the basis for concluding whether the Westinghouse minimum containment pressure analysis satisfies the following requirements:

- Section I.D.2 of Appendix K to 10 CFR Part 50, which requires that the containment pressure used in ECCS reflood calculations not exceed a pressure calculated conservatively for that purpose

- 10 CFR 50.46, which requires that ECCS cooling performance be calculated in accordance with an acceptable evaluation model

DCD Tier 2, Section 6.2.1.5, discusses the containment analysis used to determine the minimum backpressure for input as a boundary condition in the ECCS evaluation model. Generally, the core flooding rate of a PWR is dependent on the ability of the ECCS to displace steam generated in the reactor vessel; there is a direct correlation between the containment pressure and the rate of core reflood. Minimizing the containment pressure used as a boundary condition in the ECCS analysis is therefore considered conservative. Any pressurization of the containment above 101 kPa (14.7 psia) will enhance the calculated ECCS performance of the AP1000 limiting case, large-break LOCA presented in DCD Tier 2, Section 15.6.5.

DCD Tier 2, Figure 6.2.1.5-1, graphically depicts the calculated containment backpressure used by Westinghouse for the AP1000 ECCS analysis. The peak minimized containment pressure is approximately 262.69 kPa (23.4 psig), as compared to the peak pressure of approximately 508.12 kPa (59 psig) calculated for containment design and leakage considerations.

As discussed in DCD Tier 2, Section 6.2.1.5, a single-node WGOTHIC model was used to calculate the minimum containment pressure. Conditions used to minimize the calculated containment pressure were as follows:

- initial pressure of 101 kPa (14.7 psia)
- initial temperature of 32 °C (90 °F)
- initial relative humidity of 99 percent
- assumed temperature of −18 °C (0 °F) in the shield building annulus
- addition of 10 percent to the containment volume
- increase of passive heat sink surface areas by a factor of 2.1
- during the blowdown period inside containment, use of the Tagami heat transfer correlation with a multiplier of 4
- for the postblowdown period inside containment, use of the Uchida heat transfer correlation with a multiplier of 1.2
- containment purge was assumed to be in operation through two, 38.1-cm- (15-in.-) diameter lines (40.6 -cm- (16-in.-) schedule 40 pipe) until the lines are isolated at 22 seconds following the beginning of the LOCA, at a 156.5-kPa (8-psig) closure setpoint

These assumptions are consistent with those outlined in BTP Containment Systems Branch (CSB) B 6-1, “Minimum Containment Pressure Model For PWR ECCS Performance Evaluation” of SRP Section 6.2.1.5. The mass and energy releases used in the minimum containment pressure analysis were determined consistent with Appendix K to 10 CFR Part 50, and are described in WCAP-14171 (WCOBRA/TRAC). These mass and energy releases are consistent with SRP Section 6.2.1.5, which specifies that the releases should be based on Appendix K of 10 CFR Part 50.

BTP CSB 6-1 also states that the mixing of subcooled ECCS water from the break with the steam atmosphere should be assumed to minimize the pressure. In the Westinghouse analyses, the mass and energy released from the break during blowdown is assumed to mix with the containment atmosphere. Spillage of ECCS water into the containment is not modeled because all ECCS injection is directly into the vessel and no line exists from which it could spill.

In addition, BTP CSB 6-1 specifies that pressure-reducing equipment, such as containment sprays and containment fan coolers, should be assumed to be running to minimize the containment pressure. The Westinghouse minimum backpressure analysis does not assume the containment recirculation cooling system to be operating. At about 6 seconds following the initiation of the accident, the containment recirculation cooling system would be secured on a containment isolation signal, and the impact of operation of the cooling system for 6 seconds would be small. Because the breakflow is dominated by critical flow during the period when the peak clad temperature occurs, a lower containment pressure would have no effect on the RCS or cladding temperature. Therefore, the staff finds the licensing analyses without the containment recirculation cooling system to be acceptable for the AP1000 minimum containment pressure evaluation.

PCS flow is not modeled because the time period of interest in the analysis is approximately the first 150 seconds after a LOCA. During this time, the containment shell would not have heated up enough to significantly affect the containment pressure. Prior to actuation of the fourth stage of the ADS there is limited communication between the containment and the RCS, and the fourth-stage ADS valves are adequately sized and are not sensitive to containment pressure.

In conclusion, the staff finds that Westinghouse has satisfied that part of Appendix K to 10 CFR Part 50 which requires a conservative backpressure to be used in ECCS reflood calculations. Westinghouse has also satisfied, in part, 10 CFR 50.46, inasmuch as the analysis used to calculate the containment backpressure is acceptable. In particular, Westinghouse has performed its minimum containment backpressure analysis, using assumptions that minimize the calculated backpressure and which are consistent with those assumptions acceptable to the staff, by following the guidance given in BTP CSB 6-1 of SRP Section 6.2.1.5. Furthermore, Westinghouse has followed the guidance given in SRP Section 6.2.1.5 regarding the mass and energy releases. Therefore, these releases are acceptable on the basis of the staff's findings in Section 15.2.6 of this report.

Westinghouse presented the mass and energy releases to the containment during the blowdown and reflood portions of the limiting, double-ended cold-leg break transient in DCD Tier 2, Table 6.2.1.5-1, as computed by the WCOBRA/TRAC code. The staff reviewed the application of this methodology to the AP1000.

On the basis of the aforementioned considerations, the staff finds the minimum containment backpressure analysis to be acceptable. The acceptability of the credited backpressure has been evaluated in the overall context of the ECCS performance capability studies. Section 15.2.6.5 of this report provides the staff's evaluation of the ECCS performance.

6.2.1.6 Testing and Inspection

Westinghouse summarizes the functional testing and inspection of the containment vessel in DCD Tier 2, Section 6.2.1.6. DCD Tier 2, Section 3.8.2.7, describes the testing and inservice inspection of the containment vessel, while DCD Tier 2, Section 6.2.3, describes isolation testing and DCD Tier 2, Section 6.2.5, describes leak testing. The valves of the PCS are periodically stroke tested, and DCD Tier 2, Section 6.2.2.4, provides a description of the testing and inspection. Testing and inspection will be consistent with regulatory requirements and guidelines.

The baffle between the containment vessel and the shield building is equipped with removable panels and clear observation panels to allow for inspection of the containment surface. DCD Tier 2, Section 3.8.2.7 provides the requirements for inservice inspection of the steel containment vessel. DCD Tier 2, Section 6.2.2.4, provides a description of the testing to be performed.

Westinghouse states that testing is not required on any subcompartment vent or on the collection of condensation from the containment shell. DCD Tier 2, Section 5.2.5, discusses the collection of condensate from the containment shell and its use in leakage detection.

The PCS is designed to permit periodic testing of system readiness, as specified in the TS.

Preoperational Testing

Preoperational testing of the PCS is verified to provide adequate cooling of the containment. The flow rates are confirmed at the minimum initial tank level, at an intermediate step with all but one standpipe delivering flow, and at a final step with all but two standpipes delivering to the containment shell. The flow rates are measured utilizing the differential pressure across the orifices within each standpipe, and will be consistent with the following minimum flow rates (see DCD Tier 2, Table 6.2.2-1), which are greater than the flow rates used in the safety analyses, assuring that the safety analyses are conservative:

- 1783.3 L/min (471.1 gpm) at the minimum operating water level
- 902.4 L/min (238.4 gpm) at a level after the first standpipe is uncovered
- 696.5 L/min (184.0 gpm) at a level after the second standpipe is uncovered
- 573.1 L/min (151.4 gpm) at a level after the third standpipe is uncovered

The containment PCS water coverage fraction (wetted surface area) will also be measured at the base of the upper annulus, in addition to the measurements at the spring line. A full-flow test, using the PCS water storage tank to deliver the flow, will be performed. An additional test will be performed at a lower flow rate using the PCS recirculation pumps to deliver the flow. A throttle valve will be used to obtain the low flow rate (less than the full capacity of the PCS recirculation pumps). This flow rate will be reestablished for subsequent tests over the life of the plant using the throttle valves. These two benchmark tests will be used to develop acceptance criteria for the TSs. The full-flow condition is selected because it is the most important flow rate with respect to the peak pressure; the lower flow rate is selected to verify the wetting characteristics of the containment exterior surface at less than full-flow conditions.

The standpipe elevations are verified to be at the values specified in DCD Tier 2, Table 6.2.2-2.

The inventory within the tank is verified to provide 72 hours of operation from the minimum initial operating water level with a minimum flow rate over the duration in excess of 381.2 L/m (100.7 gpm). The flow rates are measured utilizing the differential pressure across the orifices within each standpipe.

The containment vessel exterior surface, above the 41.2-m (135-ft) elevation, is verified to be coated with an inorganic zinc coating. The containment vessel interior surface, from 2.1 m (7 ft) above the operating deck, is verified to be coated with an inorganic zinc coating (see DCD Tier 2, Section 6.1.2.1.5).

The passive containment cooling airflow path will be verified at the following locations:

- air inlets
- base of the outer annulus
- base of the inner annulus
- discharge structure

With either a temporary water supply or the passive containment cooling ancillary water storage tank connected to the suction of the recirculation pumps, and with either of the two pumps operating, the flow rate to the PCCWST will be in excess of 381.12 L/min (100.7 gpm), as used in the safety analyses. Temporary instrumentation or changes in the PCCWST level will be utilized to verify the flow rates. The capacity of the passive containment cooling ancillary water storage tank is verified to be adequate to supply 381.2 L/min (100.7) gpm for a duration of 4 days.

The PCCWST provides makeup water to the spent fuel pool. When aligned to the spent fuel pool, the flow rate is verified to exceed 132.5 L/min (35 gpm). Installed instrumentation will be utilized to verify the flow rate. The volume of the passive containment cooling ancillary water storage tank is verified to exceed 2,952,621.2 liters (780,000 gallons).

DCD Tier 2, Chapter 14, provides additional details for preoperational testing of the PCS; Chapter 14 of this report discusses these details.

The staff finds that the preoperational testing program, in combination with the supplemental initial test program, adequately verifies the PCS water delivery flow rates, wetted surface areas, and volume of PCS water available. These tests verify the PCS characteristics used in the licensing analyses and are acceptable. DCD Tier 2, Section 14.2.9.1.4, “Passive Containment Cooling System Testing,” describes the initial test program.

Operational Testing

Operational testing is performed to:

- Demonstrate that the sequencing of valves occurs on the initiation of Hi-2 containment pressure, and demonstrate the proper operation of remotely operated valves.

- Verify valve operation during plant operation. The normally open motor-operated valves (MOVs), in series with each normally closed air-operated isolation valve, are temporarily closed. This closing permits isolation valve stroke testing without actuation of the PCS.

- Verify waterflow delivery is consistent with the accident analysis.

- Verify visually that the path for containment cooling airflow is not obstructed by debris or foreign objects.

- Test frequency is consistent with the plant TSs (DCD Tier 2, Section 16.1, TS 3.6) and inservice testing program (DCD Tier 2, Section 3.9.6).

The operational testing program assures that the PCS is available and maintained consistent with the licensing analyses. The staff finds the operational testing program to be acceptable.

6.2.1.7 Containment Instrumentation Requirements

Instrumentation is provided to monitor the conditions inside the containment and to actuate the appropriate ESFs, should those conditions exceed the predetermined levels. As required by 10 CFR 50.34(f)(2)(xvii), instrumentation must be provided to measure, record, and provide readout in the control room of the following system parameters:

- containment pressure
- containment water level
- containment hydrogen concentration
- containment radiation intensity (high level)

In addition to these parameters, RG 1.97 recommends that instrumentation to monitor containment atmosphere and sump water temperature be provided. DCD Tier 2, Chapter 7, describes the AP1000 postaccident monitoring system considering the recommendations in RG 1.97. DCD Tier 2, Section 5.2.5, describes instrumentation to monitor RCS leakage into containment.

The containment pressure is measured by four independent pressure transmitters, and the signals are fed into the ESF actuation system, as described in DCD Tier 2, Section 7.3.1. Upon detection of high pressure inside the containment, the appropriate safety actuation signals are generated to actuate the necessary safety-related systems. If a low-pressure alarm exists, however, it does not actuate the safety-related systems.

The containment atmosphere radiation level is monitored by four independent area monitors located above the operating deck inside the containment building. The measurements are continuously fed into the ESF actuation system logic. DCD Tier 2, Section 11.5, provides information on the containment area radiation monitors, while DCD Tier 2, Section 7.3, describes the ESF actuation system operation.

The hydrogen concentration monitoring subsystem (HCMS) measures the containment hydrogen concentration but it is not part of postaccident monitoring and is non-safety related.

DCD Tier 2, Sections 6.2.4 and 7.5 describe the system, and Section 6.2.5.4 of this report presents the staff's evaluation. The staff concludes in Section 6.2.5.4 of this report that the HCMS design meets the requirements of GDC 41 and 10 CFR 50.44 as well as the provisions of draft RG 1.7, Revision 3.

DCD Tier 2, Table 7.5-1, "Post-Accident Monitoring System," contains the instrumentation provided to meet the guidance of RG 1.97. DCD Tier 2, Table 7.5-1, includes instrumentation capable of monitoring the atmospheric temperature of containment and the containment sump's water level and temperature in a harsh environment. Containment temperature is measured from 0–204 °C (32–400 °F). Containment water level can be monitored from the 21.95 m (72 ft) elevation to the 33.53 m (110 ft) elevation. The staff concluded that containment cooling status can be determined through an alternative means to direct reading of containment sump water temperature. The alternative means include either Category 2 PRHR HX inlet or outlet temperature. In the AP1000, containment sump water temperature is monitored as a Category 2 variable from 10–260 °C (50–500 °F) at the PRHR HX outlet.

The containment instrumentation described above has been designed to meet the guidance of Item II.F.1 of NUREG-0737 and RG 1.97. The staff concludes that this instrumentation meets the regulations and standards in SRP Section 6.2.1.1.A-I.G and 10 CFR 50.34(f)(2)(xvii).

6.2.1.8 Adequacy of IRWST and Containment Recirculation Screen Performance

DCD Tier 2, Section 6.3, provides information concerning the operation of the AP1000 PXS, which includes a description of the design features of the system's debris screens. The AP1000 has two sets of screens, the IRWST screens and the containment recirculation screens. DCD Tier 2, Section 6.3.2.2.7, includes a description of these screens, their design criteria, and their conformance with Revision 2 of RG 1.82, "Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident." As discussed in this section, the staff reviewed the AP1000 debris screens in accordance with the current state of knowledge concerning the issues associated with Generic Safety Issue (GSI) 191, "Assessment of Debris Accumulation on PWR Sump Performance." The NRC staff issued RAIs concerning the design adequacy of the IRWST and containment recirculation screens in a letter to the applicant dated January 21, 2003. The applicant submitted responses to the staff's RAIs as described in the evaluation below.

6.2.1.8.1 Post-LOCA Debris Generation and Washdown Potential

As the IRWST and containment recirculation screens are designed to accommodate only modest debris loadings, the AP1000 design relies heavily upon limiting the introduction of potential debris sources into containment and impeding debris transport to prevent unacceptably large debris loadings. A predominate source of postaccident debris in many reactor designs is the fibrous thermal insulation on piping and components of the RCS and other associated and colocated systems. In order to limit the challenge to the IRWST and containment recirculation screens from insulation debris, the applicant stated in DCD Tier 2, Section 6.3.2.2.7.1, that fibrous insulation will not be used in zones of the AP1000 containment where it would be vulnerable to damage by jet impingement from postulated pipe breaks.

DCD Tier 2, Section 6.3.2.2.7.1, originally defined the zones considered by the applicant to be vulnerable to damage by jet impingement in the following way:

> Insulation located in a spherical region within a distance equal to 12 inside diameters of the LOCA pipe break is assumed to be affected by the LOCA when there are intervening components, supports, structures, or other objects. In the absence of intervening components, supports, structures, or other objects, insulation in a cylindrical area extending out a distance equal to 45 inside diameters from the break along an axis that is a continuation of the pipe axis and up to 5 inside diameters in the radial direction from the axis is assumed to be affected by the LOCA.

The boundaries of these zones, from which fibrous material would be excluded, were based on calculations performed for the NRC staff by Science and Engineering Associates, Inc. (SEA), and data taken from tests performed by the Boiling Water Reactor Owners' Group (BWROG) and described in its utility resolution guidance report, NEDO-32686. In regions of containment where there are no intervening structures, the SEA calculations and BWROG tests show that fibrous insulation can be degraded into readily transportable pieces up to distances equivalent to 45 times the inner diameter of the ruptured pipe. Because the applicant's definition of the vulnerability zone for regions of containment that do not contain intervening materials is consistent with the testing and analysis described in this paragraph, the NRC staff finds it to be acceptable.

For containment regions in which jet impingement will be reflected and attenuated by intervening structures, the staff has previously considered a spherical jet impingement model to be a reasonable approximation for estimating a volume of generated debris. The NRC safety evaluation report (SER) on the BWROG's report NEDO-32686 states that a spherical impingement model appears logical for congested zones of containment, and that it may be the best approximation for estimating the amount of debris in congested zones. However, the SER also indicates that the precision of the spherical model is unsupported by either analytical modeling or experimental evidence.

Consistent with the SER on NEDO-32686, the NRC staff considers the spherical jet impingement model to have limited applicability for the AP1000. Specifically, the NRC staff agreed that systematically excluding fibrous insulation from spherical volumes (with a radius equal to 12 inside pipe diameters) surrounding postulated break locations will greatly minimize the amount of debris generated from fibrous insulation. However, the staff was unable to conclude that the applicant's controls regarding fibrous insulation will ensure that no debris would be generated from fibrous insulation by breaks in congested zones of containment.

As demonstrated in the citation above, DCD Tier 2, Section 6.3.2.2.7.1, models containment congestion as an all-or-nothing condition. It is unclear to the staff that such a binary model is capable of accurately predicting jet impingement for break locations with only mild or directional structural congestion. Under these conditions, for example, the shape of the jet impingement could resemble partially obstructed opposing cones that extend beyond the spherical boundary assumed in the DCD. Additionally, uncertainty exists relative to the spherical impingement model, even in areas of high structural congestion, because of possible variations in parameters, such as the offsets of ruptured pipes and the degree of intervening material present in the various directions about a pipe break. Thus, the staff expects that the zones actually affected by jet impingement would not be precisely spherical and concludes that portions of actual jet impingement boundaries could exceed 12 pipe diameters, even in the presence of intervening structures. For this reason, in the DSER, the staff concluded that the applicant had not sufficiently demonstrated that actual jet impingement zones, in the presence of intervening structures, would not result in the generation of debris from fibrous insulation that is located beyond a 12-pipe-diameter sphere. This was Open Item 6.2.1.8.1-1 in the DSER .

In a July 3, 2003, letter, the applicant provided a revised DCD Tier 2, Section 6.3.2.2.7.1, in which it increased the DCD spherical zone of damage from 12- to 20-pipe inside diameters. In addition, Westinghouse provided information that states that the only place where RCPB piping is located within 45-pipe inside diameters of fibrous insulation is near the top of the pressurizer. Here, chilled water lines are located at least 24-pipe inside diameters away from the RCPB piping. Because the applicant increased the spherical zone of damage from 12- to 20-pipe inside diameters and provided information that shows that, in the presence of intervening structures, actual jet impingement zones would not result in the generation of debris from fibrous insulation that is located beyond a 20-pipe-diameter sphere, the staff considers DSER Open Item 6.2.1.8.1-1 closed.

In zones vulnerable to jet impingement, the DCD states that the AP1000 will use reflective metallic insulation (RMI), or an equivalent material that will not be damaged by jet impingement or be transported to the containment recirculation screens. Testing sponsored by the NRC and the BWROG in the resolution of the boiling-water reactor (BWR) strainer blockage issue shows that the deployment of RMI within zones vulnerable to jet impingement will significantly reduce the likelihood of screen blockage in comparison to fibrous insulation. As compared to fibrous insulation, RMI is generally (1) more resistant to damage from jet impingement, (2) more difficult to transport to the debris screens, (3) less capable of accumulating uniformly on the screens, and (4) not known to interact with particulate debris in the same way that fibrous debris does (the so-called “thin-bed” effect) to result in a severe head loss across the screens.

As a result of the deployment of RMI (or an equivalent material) in zones vulnerable to jet impingement, DCD Tier 2, Section 6.3.2.2.7.1, states that, “fibrous debris is not generated by loss-of-coolant accidents.” In regard to this statement, the staff issued RAIs 650.002, 650.003, and 650.004, which questioned whether the applicant had considered all potential sources of fibrous debris that could be present in the AP1000 containment in the design of the IRWST and recirculation screens. The staff’s RAIs pointed out that, in addition to insulation, other sources of fibrous debris could be installed in containment (such as fire barriers), and that resident fibrous debris may exist in the form of dust on surfaces inside containment or as material settled onto the floor of the IRWST. To resolve the staff’s concerns regarding these potential sources of fibrous debris, in a letter dated February 21, 2003, the applicant submitted analyses of the IRWST screens in RAI 650.004 and the containment recirculation screens in RAI 650.005 to demonstrate their capability to accommodate anticipated amounts of fibrous materials. The applicant’s analyses are evaluated subsequently. Because the applicant’s analyses of the AP1000 debris screens include debris from resident fibrous material, the staff considers RAI 650.002 (which concerned resident fibrous dust) to be closed. The staff will evaluate the adequacy of the applicant's treatment of the resident fiber concern in conjunction with RAIs 650.004 and 650.005.

The staff issued RAI 650.003 to determine whether the applicant had considered fire barriers as a potential source of postaccident debris. In a letter dated February 21, 2003, the applicant responded to RAI 650.003 by stating that the fire barriers intended for use in the AP1000 containment are made of steel plates or a steel-composite material. The applicant stated that no fibrous debris would be generated from this material, and that any debris formed would either be maintained within the steel plates or would have sufficient density to sink rapidly in water. After a teleconference with the staff on April 3, 2003, the applicant agreed to revise its response to this RAI to clarify that the prohibition on fibrous materials in zones vulnerable to jet impingement and containment flooding applies not only to fibrous insulation, but also to other installed sources of fibrous material (e.g., fire barriers and ventilation filters). In a letter dated April 9, 2003, the applicant confirmed the applicability of this prohibition to other installed sources of fibrous material by submitting a revised response to RAI 650.003 and revising DCD Tier 2, Section 6.3.2.2.7.1, appropriately. Because the applicant provided the additional information requested by the staff and revised the DCD to reflect its commitment to assure that fibrous material installed in containment will not become debris that could adversely affect the IRWST and containment recirculation screens, the staff considers RAI 650.003 closed.

Similar to its position concerning the AP600, the applicant maintained that coatings used in containment below the operating deck do not have to be qualified as safety-related for the AP1000 because their failure will not interfere with core cooling by clogging the IRWST and containment recirculation screens. As discussed in DCD Tier 2, Section 6.1.2.1.6, the non-safety-related coatings used in the containment will be procured (but not applied, inspected, or monitored) according to the quality assurance requirements of Appendix B to 10 CFR Part 50. On this basis, the applicant stated in its letter dated February 21, 2003, that the non-safety-related coatings are not expected to fail. Although it does not altogether disagree with this position, the staff has historically considered at least a partial failure of non-safety-related coatings to be credible, and included the failure of coatings in its evaluation (in Sections 6.2.1.8.2 and 6.2.1.8.3 of this report) of the adequacy of the IRWST and containment recirculation screens.

DCD Tier 2, Section 6.3.8.1, states that, "Combined License applicants referencing the AP1000 will address preparation of a program to limit the amount of debris that might be left in the containment following refueling and maintenance outages." Because a significant fraction of the debris (particularly with respect to fibrous debris) that eventually reaches the debris screens could be resident debris, the staff believes that a robust containment cleanup and foreign-material control program is essential to ensuring adequate performance of the AP1000 PXS. This program is addressed by COL Action Item 6.2.1.8.1-1.

The AP1000 design includes a non-safety-related containment spray system that will be used only in the case of a severe accident. Containment spray is capable of washing down debris that might not otherwise be transported to the containment pool and IRWST. However, if a severe accident has occurred, by definition, core heat removal or coolant has already been lost, and the containment spray's effect in transporting additional debris is not significant. Therefore, in comparison to operating PWRs, the fraction of debris washed down to the IRWST and containment pool is expected to be reduced.

As a result of the applicant's design controls to limit quantities of potential debris sources in containment, particularly in regard to sources of fibrous material, the staff concludes that the amount of debris generated for the AP1000 would be small compared to most operating PWRs. In addition, much of the debris that would be generated is known not to contribute significantly to head loss under conditions applicable to the AP1000. The following two sections provide the staff's assessment of the ability of the IRWST and containment recirculation screens to accommodate anticipated quantities of postaccident debris.

6.2.1.8.2 Pool Transport and Head Loss Evaluation of the IRWST Screens

DCD Tier 2, Section 6.3.2.2.7.2, originally described the IRWST screens as being flat, vertical screens, each 6.5 m^2 (70 ft^2) in area. The two screens are located at opposite ends of the IRWST, near the bottom of the tank. The screens are described as being designed to intercept debris larger than 0.3175 cm (0.125 in.), thereby preventing it from entering the RCS. The IRWST screens are each protected by a trash rack, which is designed to prevent large debris from reaching the fine screens. A debris curb at the base of the IRWST screens is designed to prevent high-density debris from being swept along the floor of the IRWST and upward onto the screen.

During normal operation, it is expected to be difficult for debris to enter the IRWST because normally closed louvers cover its vents and overflows from the containment atmosphere. In addition, the IRWST is constructed from stainless steel and will not generate the corrosion products that contributed to strainer plugging in the carbon steel suppression pools of operating BWRs. TS Surveillance Requirement (SR) 3.5.6.8 requires a visual inspection of the IRWST screens every 24 months to ensure that they are not restricted by debris. TS SR 3.5.4.7 requires a similar 24-month inspection of the IRWST gutters, which are covered by a trash rack and are part of the containment water long-term return and recirculation system. During accident conditions, limited quantities of debris may be introduced into the IRWST (e.g., through entrainment in the condensate washdown collected by the IRWST gutters). An example of a potential debris source cited in DCD Tier 2, Section 6.3.2.2.7.2, is the inorganic zinc coating applied to the inside surface of the containment shell. (However, the DCD states that, should any coating debris enter the tank, it would tend to settle onto the tank floor by virtue of its density.) Based on the limited potential for debris generation discussed previously and the limited availability of debris that would have the potential to wash down into the IRWST, the amount of debris introduced is expected to be relatively small.

In RAI 650.004, the staff requested additional information concerning the potential for debris to be concentrated in the IRWST when the tank inventory is cycled during refueling outages. Any debris entrained in water entering the IRWST would settle out during long stagnation periods during the operating cycle, and could later become stirred up during an accident condition when the ADS is actuated. In its response to RAI 650.004, dated February 21, 2003, the applicant stated that purification processes associated with refueling activities would limit the amount of debris that would be capable of settling out on the IRWST floor. The applicant further provided qualitative reasons to support its contention that, "any resident debris that has settled on the

IRWST floor prior to an accident is not likely to be stirred up by the ADS . . . ." Reasons cited by the applicant included the subcooled state of the IRWST inventory, the location of the ADS spargers 4.87 m (16 ft) above the IRWST floor and only on one side of the tank, and the sequencing of the ADS valves. Because the applicant did not provide a quantitative analysis of the turbulence conditions within the IRWST during an ADS actuation, the staff lacks a sound basis to conclude whether the relatively small velocities needed to entrain settled debris would not be exceeded. Although the staff concurs that large amounts of debris (i.e., quantities comparable to those found in a BWR suppression pool) are unlikely to be settled on the floor of the IRWST, without a flow analysis, the staff finds that the design of the IRWST screens should include the capability to accommodate resuspension of available quantities of debris settled onto the IRWST floor.

To address this issue, the applicant's February 21, 2003, response to RAI 650.004 also included an analysis of the IRWST screens' capability to accommodate debris accumulation. The staff's review of the applicant's analysis showed that the mass of resident debris assumed by the applicant (i.e., 227 kg (500 lb)) was consistent with estimates made for current generation PWRs in the GSI 191 parametric study (NUREG/CR-6772). However, the staff could not accept this analysis, primarily because the applicant assumed that a single-density value is valid for all density-dependent calculations involving resident fibrous debris. According to the physical properties of analyzed types of fibrous materials, potentially different density values may be required to correctly determine the settling velocity (i.e., the material density), to calculate a volume from the assumed mass (i.e., the "as-found" density), and to determine the thickness and porosity of the associated debris bed (i.e., the rubblized density). As a result of the applicant's single-density assumption, which deviated significantly from the material properties of the low-density fiberglass on which the head loss data referenced by the applicant were based, the NRC staff concluded that the calculation was unacceptable. During a teleconference on April 3, 2003, the applicant agreed to resubmit its response to RAI 650.004, in light of the staff's concern. Pending an acceptable resolution of this concern, the staff considered the capability of the AP1000 IRWST screens to accommodate anticipated debris loadings to be Open Item 6.2.1.8.2-1 in the DSER.

To support closure of the open item, the applicant described in its January 13, 2004, letter, a design change to the IRWST screens which increased the fine screen area by at least a factor of 2 (i.e., $\geq$ 13 m^2 (140 ft^2)) by using a folded screen design. An increased screen area will allow the screen to tolerate more debris and lowers the water velocity at the screen face.

Two COL Action Items are provided to ensure adequate containment cleanliness is maintained, and that the increased IRWST surface area can accommodate anticipated debris loadings. DCD Tier 2, Section 6.3.8.1 (COL Action Item 6.2.1.8.1-1), "Containment Cleanliness Program," states the following:

> The Combined License applicants referencing the AP1000 will address preparation of a program to limit the amount of debris that might be left in the containment following refueling and maintenance outages. The cleanliness program will limit the storage of outage materials (such as temporary scaffolding and tools) inside containment during power operation consistent with [DCD Tier 2, Section] 6.3.8.2.

DCD Tier 2, Section 6.3.8.2 (COL Action Item 6.2.1.8.2-1), "Verification of Water Sources for Long-Term Recirculation Cooling Following a LOCA," states the following:

> The Combined License applicants referencing the AP1000 will perform an evaluation consistent with Regulatory Guide 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 [Tier 2, Section] 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.

Based on the increased screen size, in concert with the cleanliness program in COL Action Item 6.2.1.8.1-1, the minimal fibrous materials used in containment, and the other screen design features described in the DCD (trash racks, etc.) and discussed above, the staff considers the capability of the AP1000 IRWST screens to accommodate anticipated debris loadings to be acceptable. COL Action Item 6.2.1.8.2-1 will address any impact on the ability of the IRWST screens to accommodate anticipated debris loadings identified during the resolution of GSI 191, and that those impacts can be addressed using programmatic means. Therefore, the staff considers Open Item 6.2.1.8.2-1 closed.

6.2.1.8.3 Pool Transport and Head Loss Evaluation of the Containment Recirculation Screens

DCD Tier 2, Section 6.3.2.2.7.3, originally described the containment recirculation screens as being flat, vertical screens, each 6.5 m^2 (70 ft^2) in area. The screens are designed to intercept debris larger than 0.3175 cm (0.125 in.), thereby preventing it from entering the RCS. The screens are each protected by a trash rack, which is designed to prevent large debris from reaching the fine screens. The bottoms of the screens are elevated 0.61 m (2 ft) above the adjacent floor, which inhibits debris transport, much like a curb. The floor adjacent to the recirculation screens is at an elevation 3.5 m (11.5 ft) above the lowest elevation in containment. Each screen is protected from settling debris by a steel screen plate that extends outward 3 m (10 ft) in front of the screen, and 2.13 m (7 ft) to its side. The screen plates are specifically designed to prevent debris from the failure of protective coatings from approaching and potentially blocking the screens. TS SR 3.5.6.8 requires visual inspection of the recirculation screens every 24 months to ensure that they are not restricted by debris.

The low transport velocities of the AP1000, and the long time (i.e., up to 5 hours) before the recirculation mode of the passive core cooling system is initiated, will provide ample opportunity for dense debris to settle on the containment floor before suction is taken on the recirculation screens. Thus, it is unlikely that debris very much denser than water would reach the recirculation screens. In addition, the low transport velocities in the containment pool, in conjunction with the height of the recirculation screens, make it difficult for dense debris to reach and accumulate uniformly on the screen surface. The low flow velocities at the screen surface, which are typically an order of magnitude lower than the screen flow velocities at operating PWRs, also lead to reduced head losses. In addition, when the recirculation lines initially open, the water level in the IRWST is higher than the level in containment, and water flows from the IRWST backwards through the containment recirculation screens. This backflow tends to flush debris located on or near the recirculation screens away from the screens.

The water level at the beginning of recirculation is approximately 3 m (10 ft) above the top of the recirculation screens. Thus, any floating debris will remain clear of the screens. The recirculation piping inlet elevation is slightly above the compartment floor, which is substantially below the expected postaccident flood-up water level. This reduces the potential for air ingestion because recirculation does not initiate until the flood-up water level is well above the piping inlet.

The water level in containment following a LOCA would be sufficiently high that DCD Tier 2, Section 3.4.1.2.2.1, states that inventory from the containment pool would "flow back into the RCS via the break location . . . ." In light of this statement, the staff issued RAI 650.001 to request additional information concerning the potential for entrained debris to cause blockage at flow restrictions within the RCS, once flow begins entering through the break location after floodup (i.e., bypassing the recirculation screens). In a letter dated February 21, 2003, the applicant responded to RAI 650.001 by submitting an analysis which concluded that RMI debris is incapable of causing such blockage. Although the applicant's response partially addressed the staff's RAI, it was not complete because it did not address the potential for other sources of debris, such as fibrous debris and floatable debris, to enter the RCS through the break location and block requisite core cooling flowpaths. Pending the complete resolution of this concern, the staff considered debris blockage in the RCS to be Open Item 6.2.1.8.3-1 in the DSER.

In the applicant's Revision 1 response to RAI 650.001 dated April 24, 2003 (which was also cited as the initial response to Open Item 6.2.1.8.3-1 on June 23, 2003), the applicant expanded its evaluation of potential sources of debris to include resident fibrous and particle debris, floatable debris, and unqualified coating debris. With regard to particle and floatable debris, the applicant stated that such debris may be close to the density of water such that it would remain suspended for sufficient time to allow transport into the RCS through a pipe break that becomes flooded. The applicant estimated that the pressure drop across the debris to be about 6.9 kPa (1 psi). The applicant did not expect floatable debris to be a factor because the pressure in the RCS would exceed the containment pressure by several psi at the time that the containment water level passes the break elevation. With regard to unqualified coatings, the applicant stated that the high specific gravity (>1.3) would promote the settling of such debris. Even if some coating debris entered the RCS, the applicant stated that it is expected this debris would settle in the lower plenum of the reactor vessel.

The staff had a conference call with the applicant on June 26, 2003, to discuss this open item, as well as related Open Items 6.2.1.8.2-1, 6.2.1.8.3-2, and 6.2.1.8.3-3. The staff discussed the pressure drop calculations across the debris bed in the reactor core, as well as across the containment sump screens and IRWST screens. In addition, the staff asked the applicant why the face velocity of the water calculated at the protective grid was not consistent with the face velocity of the water calculated at the containment sump screens. The applicant committed to review its calculations and made these calculations available to the staff for audit.

The staff had an additional conference call with the applicant on July 29, 2003 to discuss the four open items referenced above. The applicant acknowledged a need to further review their calculations. The applicant submitted Revision 1 to Open Item 6.2.1.8.3-1 in a letter dated August 13, 2003, which provided the results of its revised calculation of the pressure loss across a debris bed located in the core. The revised calculation was based on a total of 226 kg (500 lb) of resident debris located inside containment, a portion of which was assumed, based on engineering judgement, to bypass the screens and enter the RCS. The calculation also used the BLOCKAGE code to calculate the head loss across the screens. The applicant also performed sensitivity studies with variations in amounts of debris transported to the screens and in the mass ratio of fiber versus particle debris. The applicant concluded that the bounding pressure loss through resident debris that might deposit on the lower core support plate or in the core would not reduce the flow to the core.

Subsequent to an audit of the calculations by the staff, the staff had a conference call with the applicant on August 19, 2003. The staff discussed the applicant's calculations and the assumptions in the BLOCKAGE code with the applicant. Subsequent to this conference call, the applicant submitted Revision 2 to Open Item 6.2.1.8.3-1 in a letter dated September 8, 2003. The applicant revised its calculations further and included additional sensitivity studies. The maximum pressure drop across the core was indicated to be less than 6.9 kPa (1 psi). In addition, the applicant proposed to Revise DCD Tier 2, Sections 6.3.8.1 and 6.3.8.2, which describe COL action items for a containment cleanliness program and verification of containment resident particulate debris characteristics.

As stated in Section 6.2.1.8.2 of this report, the staff's review of the applicant's analysis showed that the mass of resident debris assumed by the applicant (i.e., 227 kg (500 lb)) was consistent with estimates made for current generation PWRs in the GSI 191 parametric study (NUREG/CR-6772). However, the staff could not accept this analysis, primarily because the applicant assumed that a single-density value is valid for all density-dependent calculations involving resident fibrous debris. According to the physical properties of analyzed types of fibrous materials, potentially different density values may be required to correctly determine the settling velocity (i.e., the material density), to calculate a volume from the assumed mass (i.e., the "as-found" density), and to determine the thickness and porosity of the associated debris bed (i.e., the rubblized density). As a result of the applicant's single-density assumption, which deviated significantly from the material properties of the low-density fiberglass on which the head loss data referenced by the applicant were based, the NRC staff concluded that the calculation was unacceptable.

The staff notes that the applicant has committed to two COL action items which require the COL applicant to prepare a containment cleanliness program (COL Action Item 6.2.1.8.1-1) and evaluate that adequate long-term cooling is available considering the debris resulting from a LOCA in conjunction with debris that exists before a LOCA using RG 1.82, Revision 3, and subsequently approved NRC guidance (COL Action Item 6.2.1.8.2-1). The staff has found these two COL action items to be acceptable as discussed below and in Section 6.2.1.8.2 of this report.

Based on the design changes to the recirculation screens discussed in this section, in concert with the cleanliness program in COL Action Item 6.2.1.8.1-1, the minimal fibrous materials used in containment, and the other screen design features described in the DCD, the staff considers the capability of the AP1000 reactor core to accommodate anticipated debris loadings to be acceptable. Therefore, Open Item 6.2.1.8.3-1 is resolved.

In RAI 650.006, the staff questioned whether non-safety-related coatings inside the containment could disbond and subsequently block the containment recirculation screens. In a letter dated February 21, 2003, the applicant responded to RAI 650.006 by submitting calculations of the trajectories of settling paint particles to provide confidence that the particles are incapable of passing around the protective screen plate and blocking a significant fraction of the recirculation sump screen surface. The applicant's RAI response further stated that no coating debris can approach the recirculation screens without passing around the protective plates because coatings are not permitted on the surfaces inside the plates. ITAAC commitment 8.c(x) in DCD Tier 1, Table 2.2.3-4, states that the applicant will verify that the dry film density of non-safety-related coating materials is consistent with the assumed value in the settling calculation (i.e., $\geq$ 1600 kg/m^3 (100 lb/ft^3)). The particle sizes and settling rates assumed in the applicant's calculation are similar to or more conservative than those previously accepted by the staff in its review of the AP600 (NUREG-1512) and the Comanche Peak Steam Electric Station Units 1 and 2 (NUREG-0797, Supplement No. 9, dated March 1985). However, according to recent evidence that resident fibrous material may exist in containments, and considering operational experience and test data concerning coating failures, the staff considers that paint particles significantly smaller than 200 mils in diameter could become trapped in the interstitial locations of a fibrous debris bed and contribute to the blockage of the recirculation screens. Therefore, in a teleconference on April 3, 2003, the staff requested additional justification from the applicant to support the assumption that paint particles smaller than 200 mils are not a blockage concern for the containment recirculation screens. The staff considers the response to RAI 650.006 to be an open item pending the resolution of this concern. This was Open Item 6.2.1.8.3-2 in the DSER.

In an April 24, 2003, letter, the applicant provided more information on the justification behind the selection of 200 mils wide and 5 mils thick as the minimum coating debris size. This is the smallest particle size that would be trapped by the screen; smaller particles would pass through a clean screen and the fuel assemblies. The applicant acknowledged that smaller particles could be trapped in a fibrous debris bed, but stated that this potential impact is compensated by two effects. The first effect is that smaller diameter particles have a faster settling rate because the reduction in the "flutter" effect, which slows the settling of discs that have a higher diameter-to-thickness ratio. A faster settling rate will allow more particles to settle out prior to reaching the screens. The second effect is that the large protective plate over the screens will protect screens from those particles with much slower-than-expected settling rates and which support passive HX operation. The staff finds the justification behind the assumption that paint particles smaller than 200 mils are not a blockage concern for the containment recirculation screens to be acceptable because the two effects described above decrease the potential that smaller particles will become trapped in a fibrous bed on the screens. The staff considers Open Item 6.2.1.8.3-2 closed.

The staff's review found that insufficient information was available in the DCD to determine whether the containment recirculation screens are capable of tolerating anticipated postaccident debris loadings. Therefore, in RAI 650.005, the staff requested additional information from the applicant to determine the debris-blockage failure criterion of the containment recirculation screens. The applicant responded to RAI 650.005 in a letter dated February 21, 2003, by providing an analysis intended to demonstrate that the AP1000 recirculation screens could accommodate a mass of resident debris (i.e., 227 kg (500 lb)) that is equivalent to estimates made for current generation PWRs in the GSI 191 parametric study (NUREG/CR-6772). However, the staff could not accept this analysis, primarily because the applicant assumed that a single-density value is valid for all density-dependent calculations regarding resident fibrous debris. According to the physical properties of analyzed types of fibrous materials, potentially different density values may be required to correctly determine the settling velocity (i.e., the material density), to calculate a volume from the assumed mass (i.e., the as-found density), and to determine the thickness and porosity of the associated debris bed (i.e., the rubblized density). As a result of the applicant's single-density assumption, which deviated significantly from the material properties of the low-density fiberglass on which the head loss data referenced by the applicant were based, the NRC staff concluded that the calculation was unacceptable. During a teleconference on April 3, 2003, the applicant agreed to resubmit its response to RAI 650.005, in light of the staff's concern. Pending an acceptable resolution of this concern, the staff considered the capability of the AP1000 containment recirculation screens to accommodate anticipated debris loadings to be Open Item 6.2.1.8.3-3 in the DSER.

The applicant described in its January 13, 2004, letter, two design changes to the recirculation screens. The first design change increased the fine screen area by at least a factor of 2 (i.e., $\geq$ 13 m^2 (140 ft^2)) by using a folded screen design. An increased screen area will allow the screen to tolerate more debris and lowers the water velocity at the screen face. The second design change added a cross-connection pipe between the two recirculation screens. Based on the above design changes to the recirculation screens, in concert with the cleanliness program in COL Action Item 6.2.1.8.1-1, the minimal fibrous materials used in containment, and the other screen design features described in the DCD, the staff considers the capability of the AP1000 recirculation screens to accommodate anticipated debris loadings to be acceptable. The staff feels that COL Action Item 6.2.1.8.2-1 will capture any impact on the ability of the recirculation screens to accommodate anticipated debris loadings identified during the resolution of GSI 191, and that those impacts can be addressed using programmatic means. The staff considers DSER Open Item 6.2.1.8.3-3 closed.

The applicant described in its January 13, 2004, letter, a design change to the IRWST screens which increased the fine screen area by at least a factor of 2 (i.e., $\geq$ 13 m^2 (140 ft^2)) by using a folded screen design. An increased screen area will allow the screen to tolerate more debris and lowers the water velocity at the screen face.

Two COL Action Items are provided to ensure adequate containment cleanliness is maintained, and that the increased recirculation surface area can accommodate anticipated debris loadings. DCD Tier 2, Section 6.3.8.1 (COL Action Item 6.2.1.8.1-1) states the following:

> The Combined License applicants referencing the AP1000 will address preparation of a program to limit the amount of debris that might be left in the containment following refueling and maintenance outages. The cleanliness program will limit the storage of outage materials (such as temporary scaffolding and tools) inside containment during power operation consistent with [DCD Tier 2, Section] 6.3.8.2.

DCD Tier 2, Section 6.3.8.2 (COL Action Item 6.2.1.8.2-1) states the following:

> The Combined License applicants referencing the AP1000 will perform an evaluation consistent with Regulatory Guide 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 [Tier 2, Section] 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.

Based on the increased screen size, in concert with the cleanliness program in COL Action Item 6.2.1.8.1-1, the minimal fibrous materials used in containment, and the other screen design features described in the DCD (screen elevation above the adjacent floor, etc.) and discussed above, the staff considers the capability of the AP1000 recirculation screens to accommodate anticipated debris loadings to be acceptable. COL Action Item 6.2.1.8.2-1 will address any impact on the ability of the recirculation screens to accommodate anticipated debris loadings identified during the resolution of GSI 191, and that those impacts can be addressed using programmatic means. Therefore, the staff considers Open Item 6.2.1.8.2-1 closed.

During the development of Revision 3 to RG 1.82, the staff identified concerns related to additional debris that can be caused by chemical reactions in the containment. In a letter dated November 12, 2003, the staff requested that the applicant address the following chemical effects as they relate to the responses to Open Items 6.2.1.8.2-1, 6.2.1.8.3-1, and 6.2.1.8.3-3:

- To minimize potential debris caused by chemical reaction of the pool water with metals in the containment, exposure of bare metal surfaces (e.g., scaffolding) to containment cooling water through spray impingement or immersion should be minimized either by removal or by chemical-resistant protection (e.g., coatings or jackets).

- In addition to debris generated by jet forces from the pipe rupture, debris created by the resulting containment environment (thermal and chemical) should be considered in the analyses. Examples of this type of debris would be disbondment of coatings in the form of chips and particulates or formation of chemical debris (precipitants) caused by chemical reactions in the pool.

The staff considered the chemical effects described above as they relate to debris generation to be Open Item 6.2.1.8.3-4.

In its November 26, 2003, response to Open Item 6.2.1.8.3-4, the applicant explained that the AP1000 is designed such that there should not be a need for temporary scaffolding during outages. In addition, large storage areas outside of containment eliminate the need for storage of outage material in containment. The applicant revised DCD Tier 2, Section 6.3.8.1 to have the containment cleanliness program limit the storage of outage material in containment. Also, the applicant explained that its preferred approach is to use materials that do not need coatings or have permanent coatings to minimize coating disbondment.

The applicant also revised DCD Tier 2, Section 6.3.8.2 to include the evaluation of chemical debris. The staff feels that COL Action Item 6.2.1.8.2-1 will capture any impact on the ability of the affected components to accommodate anticipated debris loadings identified during the resolution of GSI 191 due to chemical effects, and that those impacts can be addressed using programmatic means. The staff considers Open Item 6.2.1.8.3-4 closed.

6.2.1.8.4 Conclusions

The staff completed its review of the adequacy of the performance of the IRWST and recirculation screen in light of anticipated postaccident debris loadings. The staff finds that debris screen performance has been acceptably addressed for the AP1000 PXS.

6.2.2 Containment Heat Removal Systems

In accordance with GDC 38, the system employed by the AP1000 to remove heat from the containment atmosphere under postulated DBA conditions is the PCS. As described in DCD Tier 2, Section 6.2.2, the purpose of the system is to prevent the containment from exceeding its design temperature and pressure, thereby maintaining containment integrity and reducing the driving force for postaccident radioactive releases to the environment. This function is accomplished in the PCS by evaporative and natural convective cooling, and to a lesser degree, by radiative heat transfer.

The PCS is a seismic Category 1, Westinghouse Class C system designed to Section III, Class 3 standards of the ASME Code, in accordance with RGs 1.26, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” and 1.29, “Seismic Design Classification.” As stated in DCD Tier 2, Section 6.2.2, the principal safety design bases of the PCS include the following:

- to maintain the containment internal pressure below the design value for 3 days following a DBA, without operator action

- to withstand a single failure of an active component, assuming the loss of all onsite or offsite power, without losing the ability to perform its intended safety function

- to design components necessary for accident mitigation to remain functional during, and to withstand the effects of, a DBA

A distinguishing feature of the PCS is that it relies on naturally occurring passive physical phenomena to perform its cooling function. After initial actuation, the system does not depend on any active components. This is in contrast to existing Westinghouse designs, which utilize containment sprays and safety-grade fan coolers to cool the containment. These existing systems make use of active components, including ac-powered pumps and fans.

The major components of the PCS are the primary containment vessel, which acts as the safety-grade interface to the ultimate heat sink, the shield building, PCCWST, the air baffle, air inlets, and air diffuser, and the water distribution system comprising a water distribution bucket and distribution weirs. Section 6.2.3 of this report discusses the design of the shield building.

PCS operation is initiated when the containment pressure exceeds the Hi-2 setpoint value. Upon actuation from a safety-grade signal, water from the PCCWST flows through redundant isolation valves and a flow control orifice to the water distribution bucket. The redundant series valves are the only active components in the system, and consist of a fail-open (fail-safe), AOV and a normally open, dc-powered, MOV. Further redundancy is achieved by providing three trains of piping from the PCCWST to the distribution bucket, such that a failure in one train will not affect system performance. The PCCWST has a usable capacity of 2,864,420 liters (756,700 gallons) and is filled with demineralized water.

The water distribution bucket serves to uniformly distribute water on the outside of the primary containment vessel. The bucket is supported from the roof of the shield building and is suspended above the primary containment. Water is delivered to the containment vessel via evenly spaced slots surrounding the top perimeter of the bucket. A system of weirs and collection troughs installed directly on the vessel is also provided to further aid in uniform water distribution. The resulting water film flows under the force of gravity over the exterior of the containment vessel and is evaporated by heat conducted through the vessel wall, thereby removing energy from the post-DBA containment atmosphere. Unevaporated water is collected by two floor drains at the upper annulus elevation, each with 100-percent capacity, and routed to storm drains.

The baffle wall of the PCS is structurally supported by the primary containment and is located between that structure and the shield building, thus defining two annular flowpaths. In the event of a DBA, heat removed from the containment atmosphere through the vessel wall heats the air in the annular flowpath adjacent to the exterior vessel wall, thereby reducing the air density. Air inlets at the top of the shield building are permanently open to the atmosphere, and provide a path for ambient air to enter the annular region between the shield building wall and baffle. The difference in air density in the two annular regions results in a natural circulation flow from the air inlets to the bottom of the baffle wall, and up past the exterior of the containment vessel. The resulting natural convective cooling of the containment vessel assists in removing heat from the post-DBA containment atmosphere. The air/water vapor mixture exits to the atmosphere through a diffuser at the top of the shield building.

In DCD Tier 2, Section 6.2.2, Westinghouse states that the air inlets and air diffuser have been designed so that any external wind effects will only aid the natural air circulation (a “wind-positive” design). Westinghouse further states that these structures have been designed to prevent against ice and snow buildup, and to prevent the introduction of foreign debris into the airflow path.

The staff addresses the ability of the PCS to perform its intended safety function in Section 6.2.1.1 of this report.

6.2.3 Shield Building Functional Design

The AP1000 containment design incorporates a shield building that comprises the structure and annulus that completely surrounds the primary containment vessel. This building is a cylindrical, reinforced concrete structure with a conical roof that supports the water storage tank and air diffuser (or chimney) of the PCS. It shares a common basemat with the primary containment and auxiliary building, and is designed as a seismic Category 1 structure, in accordance with RG 1.26. It has an inner radius of about 20 m (70 ft), a height of 83.3 m (273.25 ft), and a thickness of 0.9 m (3 ft) in the cylindrical section.

The two primary functions of the shield building during normal operation are to provide a barrier from radioactive systems and components inside containment to shield against radiological effects, and to protect the primary containment from external events, such as tornados and tornado-produced missiles. Under DBA conditions, the shield building serves as a key component of the PCS by aiding in the natural convective cooling of the containment.

The key structural features of the shield building are the cylindrical structure, roof structure, and lower, middle, and upper annulus areas. Additionally, the design includes the air inlets, inlet plenum, water storage tank, air diffuser, and air baffle, all functioning as part of the PCS, which is described in Section 6.2.2 of this chapter. The cylindrical section of the shield building acts as a major structural component for the complete nuclear island and supports the PCS water storage tank. Flooring and walls of the auxiliary building are also connected to the cylindrical section of the shield building. Section 3.8 of this report provides the staff’s evaluation of the containment and shield building.

6.2.4 Containment Isolation System

The CIS consists of isolation barriers, such as valves, blind flanges, and closed systems, and the associated instrumentation and controls required for the automatic or manual initiation of containment isolation. The purpose of the CIS is to permit the normal or postaccident passage of fluids through the containment boundary, while protecting against release of fission products to the environment that may be present in the containment atmosphere and fluids as a result of postulated accidents.

In DCD Tier 2, Section 6.2.3, Westinghouse provides a description of the CIS. The AP1000 has been designed to minimize the number of mechanical containment penetrations (including hatches). Also, a greater percentage of the penetrations are normally closed, and those that are normally open use fail-close valves for isolation.

The staff reviewed the description of the containment isolation system using the review guidance and acceptance criteria of Section 6.2.4 of the SRP. SRP Section 6.2.4 identifies the staff's review methodology and acceptance criteria for evaluating compliance with GDC related to those piping systems penetrating containment.

The staff's review encompassed the following areas specified by Section 6.2.4 of the SRP and 10 CFR 50.34(f)(2)(xiv):

- CIS design, including:
  - the number and location of isolation valves (e.g., the isolation valve arrangements, location of isolation valves with respect to the containment wall, purge and vent valve conformance to BTP CSB 6-4, and instrument line conformance to RG 1.11)
  - the actuation and control features for isolation valves
  - the normal positions of valves, and the positions valves take in the event of failures
  - the initiating variables for isolation signals, and the diversity and redundancy of isolation signals
  - the basis for selecting closure time limits for isolation valves
  - the redundancy of isolation barriers
  - the use of closed systems as isolation barrier substitutes for valves
- the protection provided for CISs against loss of function caused by missiles, pipe whip, and natural phenomena
- environmental conditions in the vicinity of CISs and equipment and their potential effect
- the mechanical engineering design criteria applied to isolation barriers and equipment
- the provisions for alerting operators of the need to isolate manually controlled isolation barriers
- the provisions for, and TS pertaining to, operability and leak rate testing of isolation barriers
- the calculation of containment atmosphere released prior to isolation valve closure for lines that provide a direct path to the environs

- containment purging/venting requirements of 10 CFR 50.34(f)(2)(xiv)

The following sections provide a discussion of the staff's findings and conclusions for each of the above review areas.

6.2.4.1 Number, Location, and Arrangement of Isolation Valves

The regulatory requirements relating to number, location, and arrangement of isolation valves serving containment piping penetrations are specified in GDC 55, “Reactor Coolant Pressure Boundary Penetrating Containment,” GDC 56, “Primary Containment Isolation,” and GDC 57, “Closed System Isolation Valves.” GDC 55 and 56 require two isolation valves, one inside and one outside containment, per penetration, and the valves must be locked closed or automatic, with the restriction that a simple check valve may not be used as an automatic valve outside containment. GDC 57, which applies to penetrations for which there is a closed system inside containment, requires one locked closed, automatic (but not simple check) or remote manual isolation valve outside containment. The staff reviewed Westinghouse's proposed use of containment isolation valves, as described in DCD Tier 2, Table 6.2.3-1, for conformance with these GDC. The staff reviewed the valve arrangement information for each penetration and confirmed that the number, location, and arrangement conform to the acceptance criteria. DCD Tier 2, Table 6.2.3-1, identifies the penetrations. Each penetration has an isolation device both inside and outside containment, except for the secondary coolant system isolation lines. The exception for SG (secondary coolant system) piping is typical of PWRs and is acceptable based on credit for use of the secondary coolant system piping as a closed system inside the containment, thereby satisfying the requirements of GDC 57.

6.2.4.2 Actuation and Control Features for Isolation Valves

An SRP provision and TMI (Item II.E.4.2) requirement, in accordance with 10 CFR Section 50.34(f)(2)(xiv)(A), requires that all nonessential systems be automatically isolated upon initiation of an appropriate containment isolation signal. Nonessential systems are generally those which are neither ESF systems nor systems which accomplish a function similar to an ESF system. However, non-ESF and non-safety-grade systems should be classified as essential, if their continued operation under postaccident conditions will improve the reliability of a safety function.

The staff reviewed the actuation and control features (e.g., automatic, manual, or remote manual) for each isolation device. All AP1000 containment penetrations will be closed during an accident, with the exception of the normal residual heat removal (RHR) lines, which are normally closed, and would be opened by operator action during the first 2 hours of an accident. The review confirmed that the other valves will be provided with locking devices and administrative controls (as defined in SRP Section 6.2.4) to ensure that they are normally closed, or will be provided with automatic closure controls. Normally closed, nonautomatic isolation valves have provisions for locking the valves in the closed position. Administrative controls, as well as the design of locking devices, verify that nonautomatic isolation valves are in the correct position during plant operation.

The actual stem position of each power-operated isolation valve, whether remote, manual, or automatic, is indicated in the control room and provided as input to the plant computer. A means for position indication for these valves is also provided locally at the valves. Automatic isolation devices are provided with reset features to prevent automatic return to the normal position when an isolation signal clears.

Isolation valves that must be operable following a DBA or SSE are powered by the Class 1E dc power system. Manual override and signal reset of isolation signals is provided for such valves. Consistent with the requirements of TMI Item II.E.4.2 (as invoked by 10 CFR 50.34(f)(2)(xiv)), the design of isolation instrumentation precludes the capability for ganged reopening of closed isolation valves. 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.

TMI Item II.E.4.2 requires that the design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action. The design bases for the AP1000 CIS (see DCD Tier 2, Section 6.2.3.5) include this requirement.

6.2.4.3 Normal and Fail Positions of Isolation Valves

The acceptance criterion in Section 6.2.4 of the SRP states that, upon loss of actuator power, automatic valves should take the position that provides greater safety. The staff reviewed the normal and fail positions of isolation devices indicated in DCD Tier 2, Table 6.2.3-1. The staff's review confirmed that nonmotor-operated automatic isolation devices fail in the closed position upon loss of power source (air or electrical power). MOVs are powered by Class 1E dc power, and fail in the as-is position. A single power system failure will not prevent closure of both isolation valves in a containment penetration. These features ensure single-failure-proof isolation capability for all penetrations that might be opened during operation.

TMI Item II.E.4.2 states that containment purge and vent valves must be verified closed at least every 31 days. The TSs assure compliance with this requirement.

6.2.4.4 Initiating Variables for Isolation, Diversity, and Redundancy of Isolation Signals

Various instrumentation signals are used for automatic initiation of containment isolation. The following ESF-grade signals initiate closure of containment isolation valves, as indicated in DCD Tier 2, Table 6.2.3-1:

- A containment isolation signal (DCD Tier 2, Section 7.3.1.2.1) is generated from any of the following monitored variables:
    - automatic or manual safeguards actuation signal
    - manual containment isolation actuation
    - manual actuation of the PCS signal

- A safeguards actuation signal (DCD Tier 2, Section 7.3.1.1) is initiated by any one of the following monitored variables:

  — low pressurizer pressure
  — low lead-lag, compensated steamline pressure
  — low reactor coolant inlet temperature
  — Hi-2 containment pressure
  — manual initiation

- A steamline isolation signal (DCD Tier 2, Section 7.3.1.2.10) is initiated by any of the following monitored parameters:

  — Hi-2 containment pressure
  — low reactor coolant inlet temperature
  — low lead-lag, compensated steamline pressure
  — high steamline pressure negative rate
  — manual initiation

- A main feedwater isolation signal (DCD Tier 2, Section 7.3.1.2.6) is generated by any of the following monitored parameters:

  — automatic or manual safeguards signal actuation
  — manual initiation
  — Hi-2 SG narrow-range level
  — Low-1 T_{AVG} with coincident P4 permissive
  — Low-2 T_{AVG} with P4 permissive

- A startup feedwater isolation signal (DCD Tier 2, Section 7.3.1.2.13) occurs as the result the following conditions:

  — low T_{COLD} in any loop
  — Hi-2 SG narrow-range water level in either SG
  — manual actuation of main feedwater isolation

- A SG blowdown isolation signal (DCD Tier 2, Section 7.3.1.2.11) is used for SG blowdown line isolation and is initiated by either of the following parameters:

  — PRHR HX alignment signal
  — Low narrow-range SG water level

- An automatic isolation of the normal residual heat removal (RNS) system containment isolation valve (DCD Tier 2, Section 7.3.1.2.20 is initiated by the following:

  — Hi-2 containment radioactivity
  — automatic or manual safeguards actuation signal which is used in conjunction with a safeguards signal and provides diversity for RNS system isolation
  — manual initiation

The isolation signal, as a result of the automatic or manual safeguards actuation, can be manually reset to block the isolation of the RNS to permit RNS operation after a safeguards signal.

- A containment air filtration system isolation signal (DCD Tier 2, Section 7.3.1.2.19) is initiated by the following:

    - automatic or manual safeguards actuation signal
    - manual actuation of containment isolation
    - manual actuation of passive containment cooling
    - Hi-1 containment radioactivity

The non-safety-grade diverse actuation system (DAS) signal (DCD Tier 2, Section 7.7.1.11) is also used for automatic containment isolation. The DAS is a non-safety-related instrumentation system that provides diverse backup to support risk goals.

RG 1.141, “Containment Isolation Provisions for Fluid Systems,” and TMI Item II.E.4.2 state that CIS designs shall have diversity in the parameters sensed for the initiation of containment isolation, in accordance with SRP Section 6.2.4, “Containment Isolation System.” The staff's review verified that the diversity requirement is met.

TMI Item II.E.4.2 states that the containment setpoint pressure that initiates containment isolation for nonessential penetrations must be reduced to the minimum compatible with normal operating conditions. It further states—

> The pressure setpoint selected should be far enough above the maximum expected pressure inside the containment during normal operation so that inadvertent containment isolation does not occur during normal operation from instrument drift or fluctuations due to the accuracy of the pressure sensor. A margin of 6.9 kPa (1 psi) above the maximum expected containment pressure should be adequate to account for instrument error. Any proposed values greater than 6.9 kPa (1 psi) will require detailed justification. Applicants for an operating license should use pressure history data from similar plants that have operated more than one year, if possible, to arrive at a minimum containment setpoint pressure.

Westinghouse indicated a containment isolation actuation pressure of less than or equal to 156.5 kPa (8 psig) for the AP1000 TSs. All applicable DBA analyses used this setpoint. However, a reviewer’s note in the AP1000 TSs states that the 156.5 kPa (8 psig) value (given in brackets) is included for reviewer information only, and that the actual setpoint for a plant will be determined using a setpoint methodology that incorporates NRC-accepted setpoint methodology. The reviewer’s note further states that the pressure setpoint should be specified as low as reasonable, without creating potential for spurious trips during normal operations, consistent with TMI Item II.E.4.2.

TMI Item II.E.4.2 states that containment purge and vent isolation valves must close on a high-radiation signal. The AP1000 containment air filter supply and exhaust isolation valves comply with this requirement for additional isolation signal diversity.

As indicated in the above discussion, the initiating variables and the diversity and redundancy of the AP1000 instrumentation provide a reliable means for automatic containment isolation for DBA conditions and meet the acceptance criteria of SRP Section 6.2.4. See Chapter 7 of this report for additional discussion of instrumentation.

6.2.4.5 Basis for Selection of Closure Time Limits

Westinghouse stated that the AP1000 isolation times will be consistent with the performance of standard valve operators, except where shorter limits are necessary. Shorter limits are required for containment vent and purge valves and MSIVs, and have been included in the AP1000 design. For valve sizes up to 30.5 cm (12 in.), the standard valve operator closure times of ANS-56.2-1976 are consistent with the 60-second criterion of Section 6.2.4 of the SRP. For larger valves, Westinghouse specified appropriate faster limits. These limits are consistent with assumptions and criteria for radiological dose analyses and ECCS analysis (reflood backpressure) assumptions. Westinghouse's proposed closure time limits are, therefore, acceptable.

6.2.4.6 Redundancy of Isolation Barriers

Section 6.2.4.1 of this report discusses the staff's review of redundancy for valved piping penetrations. The AP1000 containment design incorporates the following nonvalved penetrations for purposes other than permitting fluid passage into and out of the containment during normal or accident conditions:

- the fuel transfer tube
- three spare penetrations
- two personnel hatches
- an equipment hatch
- a maintenance hatch

In addition to the valved penetrations, DCD Tier 2, Table 6.2.3-1, also lists these penetrations.

The personnel air locks have redundant barriers, one of which may be opened while the other is closed. This permits personnel passage into and out of containment during plant operation. The barriers are interlocked to ensure that both doors are not opened simultaneously. Each door is provided with a testable seal.

For penetrations that are not expected to be opened during normal or accident conditions, a single isolation barrier (e.g., blind flange) is provided. Such penetrations include the equipment and maintenance hatches, fuel transfer tube, and spare penetrations. These single-barrier penetration closures are not subject to single-active failures during plant operation. A double-seal gasketing arrangement provides a means for testing, and is therefore acceptable.

6.2.4.7 Use of Closed Systems as Isolation Barriers

The SG secondary side, as bounded by the main steam, feedwater, and blowdown isolation valves, is a closed system inside containment. This feature eliminates the need for inboard containment isolation valves in the steam, feed, and blowdown lines because the SG tubes and tube sheet and secondary-system piping actually serve as a containment boundary. The SG piping penetrating containment (e.g., main steamlines) is, however, provided with isolation valves for the purpose of limiting the severity of reactor cooldown transients and to serve as a second isolation barrier. The isolation provisions for the closed system configuration conform to the criteria of GDC 57, which require a single isolation valve located outside containment, and are, therefore, acceptable.

6.2.4.8 Protection of Containment Isolation Systems against Loss of Function As a Result of Missiles, Pipe Whip, and Natural Phenomena

The staff confirmed that the CIS design bases include protection from missiles, pipe breaks, earthquakes, fire, internal and external flooding, ice, wind, and tornados. Other sections of this report discuss specific features and design criteria for the protection of systems, structures, and equipment from these phenomena.

6.2.4.9 Environmental Conditions in the Vicinity of Containment Isolation Components

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 has 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. Section 3.11 of this report discusses the staff's review of the environmental qualification of the AP1000 SSCs, including containment isolation equipment.

6.2.4.10 Mechanical Engineering Design Criteria Applied to the Containment Isolation System, Structure, and Components

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, including valves and the steam and feedwater system inside containment, are identified as Class B, equivalent to ANS Safety Class 2. Westinghouse has selected the appropriate mechanical design classification for the CIS.

6.2.4.11 Provisions for Alerting Operators of the Need to Actuate Manual Isolation Devices in the Event of an Accident

Manual operator action is not relied upon for closure of containment isolation devices that may be normally or intermittently open during power operation. There are no piping penetrations used for circulation of contaminated coolant outside containment during accident conditions.

6.2.4.12 Provisions for and TSs Pertaining to Operability and Leakage Rate Testing of Isolation Barriers

In order to permit periodic Type A, Type B, and Type C testing of the containment and its piping penetrations, special connections must be provided on the containment and on penetrations to permit application and measurement of test air pressure and venting of leakage air. The staff's review confirmed that test, vent, and drain connections are provided at suitable locations. Section 6.2.6 of this report provides the staff's evaluation of the AP1000 containment leakage testing program.

6.2.4.13 Calculation of Containment Atmosphere Released before Isolation Valve Closure for Lines that Provide a Direct Path to the Environs

The largest piping penetration that provides a direct path to the atmosphere is the 40.65-cm (16-in.) containment air filtration exhaust line. The isolation valves in this line are specified as having a 10-second closure time. This closure time is consistent with the assumptions and criteria for radiological dose analyses and the ECCS analysis (reflood backpressure) assumptions used in DCD Tier 2, Chapter 15. Westinghouse's proposed closure time limits are, therefore, acceptable.

6.2.4.14 TMI Item II.E.4.4, Vent/Purge Valve Positions

The bases for TS 3.6.3 indicate that the 40.65-cm (16-in.) containment air filtration valves will be opened as needed in Modes 1, 2, 3, and 4. The staff's position is that the opening of large valves that provide a direct path from the containment atmosphere to the environs should be minimized during power operation. The staff also notes that the plant design has very few safety-related items in containment that would require containment entry while at power. Therefore, venting or purging should occur infrequently. As a result, the containment vent/purge system should only be used for containment pressure control, as low as is reasonably achievable, or air quality considerations for personnel entry, or TS surveillances. TS SR 3.6.3.1 includes this restriction.

6.2.4.15 Conclusions

The staff has determined that the CIS meets the acceptance criteria of Section 6.2.4 of the SRP, including the NUREG-0737 TMI action plan items incorporated into 10 CFR 50.34.

6.2.5 Containment Combustible Gas Control

The AP1000 DSER stated the following:

> The AP1000 DCD for the control of combustible gas in containment during accidents does not comply with current regulations.
>
> The NRC has proposed major changes to 10 CFR 50.44, "Standards for Combustible Gas Control System in Light-Water-Cooled Power Reactors," and related changes to 10 CFR 50.34 and 10 CFR 52.47, along with the creation of a

> new rule, 10 CFR 50.46a (see 67 FR 50374, August 2, 2002). These proposed changes are meant to risk-inform the combustible gas control requirements, and constitute significant relaxations of the requirements. The staff plans to finalize the rule changes during 2003.
>
> The AP1000 DCD is written in anticipation of these rule changes. As such, it is not in compliance with the current, more restrictive regulations. Furthermore, until the proposed rule changes are final and effective, the staff cannot know for certain if the DCD will comply with the revised rule. Therefore, the issue of containment combustible gas control must remain open at this time. This is DSER Open Item 6.2.5-1.

Subsequent to the publication of the DSER, the NRC has revised its regulations regarding the control of combustible gas in containment. The revised regulations were published on September 16, 2003, and became effective on October 16, 2003. The NRC has extensively revised 10 CFR 50.44, “Combustible Gas Control for Nuclear Power Reactors,” made associated changes to 10 CFR 50.34 and 10 CFR 52.47, and added a new section, 10 CFR 50.46a, “Acceptance Criteria for Reactor Coolant System Venting Systems.” The revisions apply to current power reactor licensees, and consolidate combustible gas control regulations for future power reactor applicants and licensees. The revised rules eliminate the requirements for hydrogen recombiners and hydrogen purge systems and relax the requirements for hydrogen- and oxygen-monitoring equipment to make them commensurate with their risk significance.

In more detail, the NRC is retaining existing requirements for ensuring a mixed atmosphere, inerting BWR Mark I and II containments, and maintaining hydrogen control systems capable of accommodating an amount of hydrogen generated from a metal-water reaction involving 75 percent of the fuel cladding surrounding the active fuel region in BWR Mark III and PWR ice condenser containments. The NRC is eliminating the design-basis LOCA hydrogen release from 10 CFR 50.44, and consolidating the requirements for hydrogen and oxygen monitoring into 10 CFR 50.44. At the same time, it is relaxing safety classifications and licensee commitments to certain design and qualification criteria. The NRC is also relocating and rewording, without materially changing, the hydrogen control requirements in 10 CFR 50.34(f) to 10 CFR 50.44. The high point vent requirements are being relocated from 10 CFR 50.44 to a new 10 CFR 50.46a, with a change that eliminates a requirement prohibiting the venting of the RCS if it could ‘‘aggravate’’ the challenge to containment.

The staff is now able to complete its review of containment combustible gas control and close DSER Open Item 6.2.5-1.

Combustible gas within the AP1000 containment is controlled by the containment hydrogen control system. This system consists of the hydrogen ignition subsystem, the non-safety-related hydrogen recombination subsystem, and the hydrogen concentration monitoring subsystem.

The containment hydrogen control system serves the following functions:

- hydrogen concentration monitoring

- hydrogen control during and following degraded core or core melt scenarios (provided by hydrogen igniters)

In addition, two non-safety-related passive autocatalytic recombiners (PARs) are provided for defense-in-depth protection against the buildup of hydrogen following a LOCA.

The hydrogen ignition subsystem meets the requirements of 10 CFR 50.44 for future water-cooled reactors. The design must limit hydrogen concentrations in containment during and following an accident that releases an equivalent amount of hydrogen as would be generated from a 100-percent, fuel clad-coolant reaction, uniformly distributed, to less than 10 percent (by volume), and maintain containment structural integrity and appropriate accident-mitigating features. This requirement was promulgated to address the lessons learned from the TMI accident. This type of accident is considered beyond the design basis and will be referred to as the severe accident case in this section. In the severe accident case, the hydrogen generation from the fuel clad-coolant reaction could be sufficiently rapid that it may not be possible to prevent the hydrogen concentration in the containment from exceeding the lower flammability limit. The hydrogen ignition subsystem is designed to promote hydrogen burning soon after the lower flammability limit is reached in the vicinity of an igniter. Initiation of hydrogen burning at the lower level of hydrogen flammability will prevent combustion at higher hydrogen concentrations, and provides confidence that containment structural integrity can be maintained during hydrogen burns.

6.2.5.1 Hydrogen Ignition Subsystem

For severe accident hydrogen control, the AP1000 containment has been provided with 64 hydrogen igniters. The igniter assembly is designed to maintain the surface temperature within a range of 870 °C to 927 °C (1600 °F to 1700 °F) in the anticipated containment environment following a LOCA. A spray shield is provided to protect the igniter from falling water drops (resulting from condensation of steam on the containment shell and on nearby equipment and structures).

The igniters have been divided into two power groups. Power to each group will be normally provided by offsite power. However, should offsite power be unavailable, then each of the power groups is powered by one of the onsite nonessential diesels. Finally, should the diesels fail to provide power, then the non-Class 1E batteries for each group will support approximately 4 hours of igniter operation. Assignment of igniters to each group is based on at least one igniter from each group providing coverage for each compartment or area.

The hydrogen ignition subsystem has been designed to promote hydrogen burning at a low concentration. Igniters have been placed in the major regions of the containment where hydrogen may be released, through which it may flow, or where it may accumulate. DCD Tier 2, Table 6.2.4-6, provides the criteria used in the evaluation and the application of the criteria to specific compartments. DCD Tier 2, Figures 6.2.4-5 through 6.2.4-13, provide the location of igniters throughout containment. DCD Tier 2, Table 6.2.4-7, also summarizes the location of igniters, and identifies subcompartment/regions and which igniters by power group provide protection. The locations identified are considered approximations (± 1 m (2.5 ft)), with the final locations governed by the installation details.

The staff's review of the number and location of igniters focused on the major transport paths of hydrogen inside the containment to ensure that hydrogen can be burned close to the release point. One of the release paths considered was through the IRWST via the first three stages of the ADS. Two igniters are located within the IRWST below the tank roof of the IRWST and above the spargers. In the event of hydrogen releases from the spargers, the igniters directly above the release points will provide the most immediate point of recombination. In the event that the IRWST is hydrogen rich and air is drawn into the IRWST, the mixture will become flammable. To provide for this type of recombination, the two inlet vents, on the PRHR side of the IRWST, have each been fitted with an igniter. Should the environment within the IRWST be inerted or otherwise not be ignited by the assemblies above the sparger, the hydrogen can be ignited as it exhausts from the IRWST at any of four vents fitted with igniter assemblies.

Flow from the IRWST vents, located at Elevation 41.15 m (135 ft), exhausts into the upper compartment. Igniter coverage for the upper compartment includes 10 igniters at Elevations 49.4 m–53.64 m (162 ft–176 ft), 4 igniters at Elevation 69.5 m (228 ft), and 4 igniters at Elevation 78.33 m (257 ft).

Another important flowpath is through the fourth stage of the ADS which relieves, at Elevation 34.1 m (112 ft), into the SG compartments. Hydrogen flow into the SG compartment will be burned by two igniters at Elevation 36.58 m (120 ft) and 2 igniters at Elevation 42.37 m (139 ft). Hydrogen leaving the SG compartment is burned in the upper compartment. This flowpath would also apply to hydrogen released through any RCS break in the SG compartment.

Finally, the staff verified that the 15 major regions or compartments identified by Westinghouse in DCD Tier 2, Tables 6.2.4-6 and 6.2.4-7 had at least two igniters, and they included the enclosed areas within containment. Two enclosed areas, the reactor cavity and the north chemical and volume control system (CVS) equipment room, do not have igniter coverage or do not have igniters directly over the RCS piping. Hydrogen releases within the reactor cavity will flow either through the vertical access tunnel, through the opening around the RCS hot- and cold-legs into the loop compartments, or, if the refueling cavity seal ring fails, then potentially through the refueling cavity. Each of these adjacent regions or compartments has at least four igniters. The staff concludes that igniter coverage of the reactor cavity is not required because (1) the reactor cavity would most likely be flooded either through the break or by the cavity flooding system, (2) adequate igniter coverage is available in hydrogen pathways from the reactor cavity, and (3) any maintenance or inspection would result in elevated personnel exposure.

Although igniters have not been located directly over the RCS piping in the north CVS room, two igniters have been located near the ceiling of the equipment room between the equipment module and the major relief paths from the compartment. The staff finds this exception from the igniter location criteria in DCD Tier 2, Table 6.2.4-6 of DCD Tier 2 to be acceptable.

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.

An additional consideration is the potential of generating significant concentration gradients within the containment during the course of the event. The staff does not expect significant stratification within the AP1000 containment based on the containment-mixing evaluation (below) and the number and location of igniters provided for the AP1000 containment.

The hydrogen ignition subsystem has been identified as one of the systems to be included in the equipment survivability program. DCD Tier 2, Appendix 19D and Appendix D to the AP1000 PRA discuss equipment survivability; Section 19.2.3.3.7 of this report evaluates this feature.

The hydrogen ignition subsystem conforms to the requirements of 10 CFR 50.44 by providing reasonable assurance that uniformly distributed hydrogen concentrations inside containment will not exceed 10 percent by volume.

6.2.5.2 Hydrogen Recombination Subsystem

The staff used the requirements of 10 CFR 50.44 to review the hydrogen recombination subsystem in the AP1000 design.

The hydrogen recombination subsystem, in the AP1000 design, serves no safety-related function, and therefore, has no nuclear safety design basis. The subsystem consists of two non-safety-related PARs installed inside the containment above the operating deck at approximate elevations of 49.4 m (162 ft) and 50.6 m (166 ft) respectively, each about 3.96 m (13 ft) inboard from the containment shell. The PARs recombine hydrogen and oxygen in the containment atmosphere to make water at a rate too slow to remove hydrogen from the containment atmosphere quickly enough to be of significant benefit during a severe accident. The PARs are provided for defense-in-depth protection against the buildup of hydrogen following a LOCA.

Based on its review, the staff finds that the hydrogen recombination subsystem is a non-safety-related system and serves no safety-related function, and its failure does not lead to the failure of any safety systems. The staff, therefore, concludes that the requirements of 10 CFR 50.44 are met, because hydrogen recombiners or similar systems are not required to control combustible gases during a design-basis LOCA. Based on the above, and the fact that its failure will not prevent safe shutdown, the staff finds the hydrogen recombination subsystem to be acceptable.

6.2.5.3 Containment Atmosphere Mixing

Another requirement of 10 CFR 50.44 for future water-cooled reactors is that all containments must have a capability for ensuring a mixed atmosphere during design-basis and significant beyond-design-basis accidents.

In meeting the requirements of 10 CFR 50.44 to provide the capability for ensuring a mixed atmosphere in the containment, and the requirements of GDC 41 to provide systems as necessary to ensure that containment integrity is maintained, a system (active, passive, or a combination of the two) should be provided to mix the combustible gases within the containment during design-basis and significant beyond-design-basis accidents. An analysis should be presented that demonstrates that excessive stratification of combustible gases will not occur within the containment or within a containment subcompartment. The containment internal structures should have design features that promote the free circulation of the atmosphere. An analysis of the effectiveness of these features for convective mixing should be presented. This analysis is acceptable if it shows that combustible gases will not accumulate within a compartment or cubicle to a level that supports combustion or detonation which could cause loss of containment integrity. As discussed below, the applicant has done this for the AP1000.

The AP1000 relies on natural circulation currents enhanced by the passive containment cooling system (PCS) to inhibit stratification of the containment atmosphere. DCD Tier 2, Appendix 6A discusses the physical mechanisms of natural circulation mixing that occur in the AP1000. Steam generated by decay heat can vent into the containment atmosphere in the form of a jet plume through the postulated break or the fourth stage of the ADS. The interaction of the plume with the ambient atmosphere can be described in terms of entrainment flow induced by the plume. Entrainment flow results in the mixing of ambient atmosphere with the steamflow in the plume. The plume will rise to the containment dome where the steam will be condensed on the inner surface of the containment shell, and the resulting cooler, denser air will fall to the operating deck.

Westinghouse provided an estimate of the degree of mixing by calculating the volumetric flow rates of gas entrained by a rising buoyant plume associated with steam generated by decay heat. The calculations were made on the basis of a steam production rate corresponding to decay heat at 1 hour and 24 hours into the accident. Entrainment flow rates were calculated using equations presented in an article by Peterson in Volume 37, Supplement 1, of the International Journal of Heat and Mass Transfer, entitled, “Scaling and Analysis of Mixing in Large Stratified Volumes.” In the Westinghouse estimate, no credit was taken for cold plumes falling from the containment dome, which causes further circulation above the operating deck. Westinghouse estimated the circulation time constant at 1 hour to be 340 seconds, and at 24 hours to be 462 seconds. The staff performed confirmatory calculations of this methodology during its AP600 review using the same equations as Westinghouse, but with containment atmospheric conditions calculated by the staff, which indicated that the estimates were reasonable. The staff, therefore, is assured that the AP1000 estimates are also reasonable.

Westinghouse has arranged containment structures to promote mixing via natural circulation. Two general characteristics have been incorporated into the design of the AP1000 to promote mixing and eliminate dead-end compartments. The compartments below deck are large open volumes with relatively large interconnections, which promote mixing throughout the below-deck region. All compartments below deck are provided with openings through the top of the compartment to eliminate the potential for a dead pocket of high hydrogen concentration.

The accumulator and CVS compartments and the reactor cavity, including the reactor coolant drain tank room, do not participate in the natural circulation flow because they are dead-ended or filled with water. The IRWST compartment is essentially sealed at the vents by flappers after blowdown. The CVS and IRWST compartments are included as confined volumes that may have water pools that provide a source of hydrogen. However, each volume has igniters which will prevent excessive hydrogen buildup. The other compartments are either completely water filled or do not contain a significant pool of water for hydrogen generation. The staff finds that these compartments will not accumulate combustible gases to a level that supports combustion or detonation, which could cause loss of containment integrity.

In its analysis, Westinghouse assumed that the fission products and hydrogen released to the containment following a postulated design-basis LOCA are homogeneously distributed in the containment atmosphere within the open compartments that participate in natural circulation. The staff finds this assumption to be reasonable. This finding is based on (1) the ability of the PCS to enhance the condensation of steam and the entrainment of air inside containment, (2) analyses performed by Westinghouse, using a method confirmed by the staff, which show that the containment atmosphere above the operating deck is recirculated approximately every 8 minutes, 24 hours after a LOCA, (3) containment structures that have been arranged to promote mixing by means of natural circulation, and (4) CVS and IRWST compartments that have been provided with igniters.

6.2.5.4 Hydrogen Concentration Monitoring Subsystem

To satisfy the design requirements of GDC 41, combustible gas control system designs should include instrumentation needed to monitor system or component performance under normal or accident conditions. As stated in 10 CFR 50.44(c)(4)(ii), the equipment for monitoring hydrogen must be functional, reliable, and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond-design-basis accident for accident management, including emergency planning.

Draft RG 1.7, Revision 3 (ADAMS Accession No. ML031670912, Attachment 5, dated July 24, 2003), says that non-safety-related, commercial-grade hydrogen monitors can be used to meet these criteria if they comply with the following:

- Category 3 design and qualification criteria of RG 1.97 for monitors used as diagnostic or backup indicators

- Category 2 power source design and qualification criteria, as specified in Table 1 of RG 1.97

Table 1 of RG 1.97, Revision 3, dated May 1983, states that a Category 2 power source is a “high-reliability power source, not necessarily standby power.” A Category 3 power source has no specific provisions associated with it.

The HCMS, as described in DCD Tier 2, Sections 6.2.4 and 7.5, consists of three non-Class 1E sensors. The sensors are placed in the upper dome where bulk hydrogen concentration can be monitored.

Hydrogen concentration is continuously indicated in the MCR. Additionally, high-hydrogen concentration alarms are provided in the MCR. The sensors are designed to provide a rapid response detection of changes in the bulk containment hydrogen concentration, and have a measurement range from 0 to 20 percent hydrogen. The response time of the sensor is at least 90 percent in 10 seconds.

The HCMS meets the guidance of RG 1.97 for Category 3 instruments, as described in DCD Tier 2, Section 7.5, and evaluated in Section 7.5 of this report. Further, the hydrogen sensors are powered by the non-Class 1E dc and uninterruptible power supply (UPS) system, which the staff considers to be a high-reliability power source.

The equipment survivability assessment also includes the HCMS. DCD Tier 2, Appendix 19D and Appendix D to the AP1000 PRA discuss equipment survivability; Section 19.2.3.3.7 of this report evaluates this feature.

The staff concludes that the HCMS design meets the requirements of GDC 41 and 10 CFR 50.44, as well as the provisions of draft RG 1.7, Revision 3.

6.2.5.5 Conclusions

The staff has determined that the containment hydrogen control system meets the requirements of GDC 41 and 10 CFR 50.44, as well as the guidelines of draft RG 1.7, Revision 3. DSER Open Item 6.2.5-1 is closed.

6.2.6 Containment Leakage Testing

The applicant's top level description of the proposed containment leakage rate testing program for AP1000 facilities is described in DCD Tier 2, Section 6.2.5 and in the proposed TSs of DCD Tier 2, Chapter 16. The test program will conform to the requirements of 10 CFR Part 50, Appendix J. The staff reviewed the information in the DCD for conformance to 10 CFR Part 50, Appendix J, and to GDC 52, "Capability for Containment Leakage Rate Testing," GDC 53, "Provisions for Containment Testing and Inspection," and GDC 54, "Piping Systems Penetrating Containment." GDC 52 requires the containment and associated equipment to be designed such that the periodic containment integrated leakage rate tests can be conducted at containment design pressure. GDC 53 requires that the containment allow periodic inspection, surveillance, and testing of certain systems, structures, and components. GDC 54 requires piping systems penetrating containment to have leak detection, isolation, and containment capabilities having appropriate redundancy, reliability, and performance capabilities, and with the capability for periodic testing of isolation valve operability and leakage rate. The staff used the guidance, staff positions, and acceptance criteria of SRP Section 6.2.6 and RG 1.163, "Performance-Based Containment Leak-Test Program," in conducting its review.

Each COL applicant will develop a "Containment Leakage Rate Testing Program" as specified in DCD Tier 2, Section 6.2.5 and by AP1000 TS 5.5.8. This program will identify which Option of Appendix J will be implemented. Option A provides prescriptive requirements, and Option B provides performance-based requirements. This program will also identify the specific TS surveillance requirements and test criteria for containment leakage rate tests. This is COL Action Item 6.2.6-1.

The applicant based the proposed TS in DCD Tier 2, Chapter 16 on Appendix J Option B, which is the Option more likely to be chosen by a COL applicant.

The staff review of the AP1000 containment leakage rate testing program encompassed the following review areas, as identified in SRP Section 6.2.6:

- Type A (integrated) leakage rate testing, including pretest requirements, general test methods, acceptance criteria for preoperational and periodic leakage rate tests, provisions for additional testing in the event of failure to meet acceptance criteria, and scheduling of tests.

- Containment penetration local (Type B) leakage rate testing, including identification of containment penetrations, general test methods, test pressures, acceptance criteria, and scheduling of tests.

- Containment isolation valve local (Type C) leakage rate testing, including identification of isolation valves, general test methods, test pressures, acceptance criteria, and scheduling of tests.

- Proposed TSs requirements pertaining to containment leakage rate testing.

The staff's findings for each of the above areas are discussed below. See also the staff's evaluation of the ITAAC in Chapter 14 of this report.

6.2.6.1 <u>Containment Integrated Leakage Rate Type A Tests</u>

Type A tests serve to provide assurance that the containment leakage rate, in the event of an accident, will not exceed the values assumed in the analyses of the radiological consequences of DBAs. An initial preoperational Type A test will be performed prior to initial startup, and periodic Type A tests and postrepair tests will be performed thereafter.

<u>Pretest Requirements for Type A Tests</u>

The DCD confirms that each Type A test will include the following pretest actions:

- A general containment inspection (internal and external) will be conducted of accessible areas. Any structural deformation or structural deterioration will be repaired before the Type A test; otherwise, the Type A test will be conducted in an as-found condition (i.e., before maintenance on valves, gaskets, seals, etc.).

- Isolation valves will be placed in their accident position using the normal method of operation, unless placement in that position is unsafe or impractical.

- Portions of fluid systems penetrating containment, that are part of the RCS boundary and that are open to the containment atmosphere under LOCA conditions, will be vented to the containment atmosphere.

- Portions of systems inside containment that penetrate containment and could rupture under LOCA conditions will be vented to the containment atmosphere and drained of fluid (unless the system would be watersealed or operating during an accident) to expose the isolation valves to the pressurized containment atmosphere.

- Components, such as tanks and instrumentation, inside containment will be vented to the containment atmosphere or removed from the containment, as necessary, to protect them against the effects of test pressure, or to preclude leakage that could affect the accuracy of the Type A test.

- Test conditions will be allowed to stabilize for at least 4 hours before beginning the test.

Compliance with the above satisfies the pretest requirements of Appendix J.

Test Method for Type A Tests

The DCD indicates that, consistent with ANSI/ANS-56.8-1994, Type A tests will use the "absolute" method and the "mass point" method. The containment will be pressurized with clean dry air to a pressure of P_a. P_a is the calculated peak containment internal pressure for the design-basis LOCA. The accuracy of the test will be verified by a supplemental test using methodology consistent with ANSI/ANS-56.8-1994. This test methodology is in accordance with the requirements of Appendix J to 10 CFR Part 50 and the guidance of RG 1.163, "Performance-Based Containment Leak-Test Program."

A permanently installed, non-safety-related piping system will be provided to facilitate controlled pressurization and depressurization of the containment. Portable compressors will be temporarily connected to the piping system for testing.

Test Acceptance Criteria

The maximum allowable leakage rate (L_a) is 0.10 percent of the containment air weight per day at P_a. During the first startup following testing, the leakage rate acceptance criterion will be 0.75 L_a, which is in accordance with the provisions of Appendix J to 10 CFR Part 50, SRP Section 6.2.6, and RG 1.163. The allowable leakage rate of 0.10 percent per day is consistent with the value used in analyses of the radiological consequences of a LOCA, as cited in DCD Tier 2, Table 15.6.5-2, and is consistent with the provisions of Section 6.2.6 of the SRP. It is, therefore, an acceptable leakage rate.

Provisions for Additional Testing in the Event of Failure to Meet Acceptance Criteria

ANSI/ANS-56.8-1994 specifies appropriate leakage pathway isolation, repair, and adjustment criteria to assure that overall as-found and as-left measurements are accurately determined to the extent possible, and without the need for test termination and a subsequent retest. If any

Type A test fails to meet the test acceptance criteria, the test schedule for subsequent tests will be adjusted in accordance with the requirements of the containment leakage rate testing program.

Scheduling of Type A Tests

An initial preoperational Type A test will be performed before initial power operation. Periodic Type A tests will be scheduled in accordance with the containment leakage rate testing program.

6.2.6.2 Containment Penetration Leakage Rate Type B Tests

Type B tests are intended to detect or measure the leakage rate across pressure-retaining or leakage-limiting boundaries other than containment isolation valves.

Identification of Containment Penetrations

Type B penetrations incorporate features such as resilient seals, gaskets, or bellows. The following four containment penetration types will receive preoperational and periodic Type B tests:

(1) penetrations having resilient seals, gaskets, or sealant compounds
(2) air locks and associated door seals
(3) maintenance and equipment hatches and associated seals
(4) electrical penetrations

These Type B penetrations include 1 main equipment hatch, 2 personnel air locks, 1 fuel transfer tube, 1 maintenance hatch, 32 electrical penetration assemblies, and 3 spare electrical penetration assemblies.

General Test Methods

The DCD states that the test boundary will be pressurized with air or nitrogen using local test connections. The pressure decay or flow meter makeup flow rate test methods will be used for leakage rate measurement.

Test Pressures

In the DCD, Westinghouse states that the test pressure will not be less than P_a.

Acceptance Criteria

In the DCD, Westinghouse states that the Type B leakage rate test results will be combined with the Type C results, in accordance with Appendix J to 10 CFR Part 50. The combined Types B and C acceptance criterion is 0.6 L_a. In addition, air lock chambers and individual doors must meet the specific leakage rate acceptance criteria identified in the TSs.

Scheduling of Tests

The schedules for periodic Type B leak rate tests will be in accordance with the containment leakage rate testing program to be developed by each COL applicant.

6.2.6.3 Containment Isolation Valve Leakage Rate Tests

Type C tests measure containment piping penetration/isolation valve leakage rates.

Identification of Isolation Valves Subject to Type C Testing

Valves at the containment boundary in SG and associated secondary-system piping will not be Type C tested, but the closed system inside containment will be tested with the containment (i.e., during Type A testing, the SG secondary side will be vented to the atmosphere outside containment). The requirements of 10 CFR Part 50, Appendix J, Option A, paragraph II.H identify those isolation valves included under Type C testing requirements; paragraph II.H.4 requires Type C testing for the containment isolation valves in main steam, feedwater, and similar piping of boiling-water reactors, but not pressurized-water reactors. For Option B of Appendix J to 10 CFR Part 50, RG 1.163 and ANSI/ANS-56.8-1994 provide similar guidance. Since the AP1000 is a pressurized-water reactor, these valves are not required to be Type C tested. The other containment isolation valves will be Type C tested.

General Test Methods

Isolation valves whose seats may be exposed to the containment atmosphere during a LOCA will be pneumatically tested with air or nitrogen. Valves in lines that would be filled with liquid for at least 30 days during the course of a LOCA will be tested with that liquid. Isolation valves will be closed by normal means without preliminary exercising or adjustments. Piping within the test boundary will be drained as necessary to assure that a water seal does not produce inaccurate results. The pressure decay method or flow meter makeup method of leakage rate measurement will be used.

Test Pressures

The test pressure will be P_a for pneumatic tests and 1.1 P_a for liquid tests.

Acceptance Criteria

Type C test results will be combined with Type B results.

Scheduling of Tests

Type C tests will be performed periodically in accordance with the containment leakage rate testing program to be developed by each COL applicant. The staff finds that the leakage rate testing provisions proposed for Type A, Type B, and Type C testing are acceptable because they are in accordance with the requirements of Appendix J to 10 CFR Part 50 and the appropriate guidance documents cited above.

6.2.6.4 Technical Specifications

In DSER Section 6.2.6.4, "Technical Specifications," the staff found one exception to staff guidance concerning the format and content of TSs for containment leakage rate testing. The numerical value of P_a should be stated in the TS, but was not. This is inconsistent with the requirements of Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," to 10 CFR Part 50. Option B of Appendix J to 10 CFR Part 50 requires the numerical value of P_a to be specified in the TS.[1]

TS 5.5.8, "Containment Leakage Rate Testing Program," states, "The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is less than the design pressure of containment." In contrast, the Westinghouse Owners Group (WOG) Standard TSs state, "The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is [45 psig]." This apparent difference was designated as Open Item 6.2.6.4-1 in the DSER.

Westinghouse has subsequently changed the subject passage in TS 5.5.8 to state, "The calculated peak containment internal pressure for the design basis loss of coolant accident, P_a, is [57.8] psig. The containment design pressure is 59 psig."

This change resolves the staff's concern and is consistent with the WOG Standard TSs and the requirements of Appendix J to 10 CFR Part 50. Therefore, Open Item 6.2.6.4-1 is closed.

6.2.6.5 Conclusions

On the basis of its review the staff concludes that the proposed AP1000 containment leakage rate testing program complies with the acceptance criteria of Section 6.2.6 of the SRP. Compliance with the SRP acceptance criteria provides adequate assurance that containment leaktight integrity can be verified before initial operation and periodically throughout its service life. Compliance with the criteria in Section 6.2.6 of the SRP, as described in this section, constitutes an acceptable basis for satisfying the containment leakage rate testing requirements of GDC 52, GDC 53, and GDC 54, and Appendix J to 10 CFR Part 50.

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[1]Appendix J allows any applicant or licensee to choose to conform to either Option A of Appendix J (Prescriptive Requirements), Option B (Performance-Based Requirements), or a specific combination of Options A and B. The plant TS must specify which choice the applicant or licensee has made. THE WOG STS contains three versions of this TS, to account for these possibilities. Two of the versions (Option B and Options A and B combined) specify the value of P_a, but the Option A version does not. This is because Option A does not require it; Option B does. The AP1000 DCD allows COL applicants to choose which option of Appendix J they want, but the staff considers it unlikely that an applicant will choose Option A alone. All operating plants currently have chosen either Option B or a combination of Options A and B, because of the cost savings to be realized by using Option B. The AP1000 TS proposes to follow the Option B model.

6.2.7 Fracture Prevention of Containment Pressure Boundary

GDC 1 requires that SSCs important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function.

GDC 16 requires that reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

GDC 51, "Fracture Prevention of Containment Pressure Boundary," requires that the reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner, and (2) the probability of rapidly propagating fracture is minimized.

The staff reviewed the AP1000 DCD to ascertain whether containment pressure boundary materials meet the requirements of GDC 1, 16, and 51.

Summary of Technical Information

DCD Tier 2, Section 3.8.2, indicates that the AP1000 containment vessel will use SA738, Grade B, material. DCD Tier 2, Section 3.8.2, also states that the materials for the AP1000 containment vessel, including the equipment hatches, personnel locks, penetrations, attachments, and appurtenances will meet the requirements of Subsection NE-2000 of the ASME Code, Section III.

Staff Evaluation

SA738, Grade B, material is an ASME Code material that is appropriate for the intended containment vessel application. The staff finds acceptable the selection of SA-738, Grade B, material for the AP1000 containment vessel, and the design and construction in accordance with the requirements of ASME Code, Subsection NE-2000. However, the staff requested, in RAI 252.009, that the following requirements be provided to supplement the requirements of specification SA-738 and that these requirements be included in the AP1000 DCD:

- Supplementary Requirement S1.7, "Vacuum Carbon-Deoxidized Steel," of Material Specification SA-738 applies to this material

- Supplementary Requirement S20, "Maximum Carbon Equivalent for Weldability," of Material Specification SA-738 also applies to this material

These two requirements are needed to ensure adequate material properties and weldability of the containment vessel material. ASME Code, Section III, exempts SA-738, Grade B, material up to 4.44 cm (1.75 in.) of thickness from postweld stress relief heat treatment. The AP1000 containment vessel is 4.44 cm (1.75 in.) thick. Because the containment vessel material thickness is 4.44 cm (1.75 in.) thick, the welds will not be stress relieved and, therefore, higher residual stresses will be present in the welds. Also, the material will likely be procured in the quenched and tempered condition. Welding will reduce the impact properties of the material in the heat affected zone. Requiring vacuum-degassed steel will ensure adequate material properties because nonmetallic inclusions, such as oxides and silicates, will be minimized as a result of the vacuum degassing of the steel. The S20 carbon-equivalent weldability check will ensure that the steel is readily weldable.

Westinghouse responded to RAI 252.009 by revising the DCD to require that supplementary requirements S1.7 and S20 be specified for the AP1000 containment vessel material.

Conclusions

Based on the review of the information included in the AP1000 DCD and the fact that the applicant will meet the requirements of Subsection NE-2000 of the ASME Code, Section III, the staff finds that the fracture toughness of the materials of the reactor containment pressure boundary meet the fracture toughness requirements invoked for ASME Code Section III, Subsection NE, Class MC materials. This satisfies the requirements of GDC 51 for fracture prevention of the containment pressure boundary. Meeting the requirements of ASME Code, Section III, also satisfies the requirements of GDC 1 for quality standards and records and GDC 16 for containment design.

The staff, therefore, concludes that reasonable assurance will be provided that the materials of the reactor containment pressure boundary, under operating, maintenance, testing, and postulated accident conditions, will not undergo brittle fracture and that the probability of rapidly propagating fracture will be minimized, so that the requirements of GDC 1, 16, and 51 will be met.

6.2.8 In-Containment Refueling Water Storage Tank Hydrodynamic Loads

DCD Tier 2, Section 6.3.2.2.3, Table 6.3-2, and Figure 6.3-4 describe the IRWST as a stainless steel-lined tank located underneath the operating deck inside the containment. The IRWST is AP1000 Equipment Class C and is designed to meet seismic Category I requirements. The tank is constructed as an integral part of the containment internal structures and is isolated from the steel containment vessel. The tank contains a minimum water volume of 2093 m^3 (73,900 ft^3).

The AP1000 design utilizes an ADS to depressurize the RCS so that long-term gravity cooling of the RCS may be established following various postulated plant events. The ADS system is composed of four distinct stages for blowdown of the RCS; the first, second, and third stages discharge into the IRWST. These discharges enter the IRWST via two submerged spargers so that the steam/water discharge from the RCS is quenched in the IRWST water. Discharging a hot pressurized steam/water mixture into a pool of relatively cool water is an efficient method for quenching the hot pressurized mixture. However, it also produces significant oscillatory hydrodynamic loads on the IRWST structure. These loads must be incorporated into the design of the IRWST structure.

To prevent imposing excessive dynamic loads on the tank structures, the spargers provide a controlled distribution of steamflow.

For the AP600, the hydrodynamic loads were determined based on tests conducted at the valve and pressure operating related experiments (VAPORE) test facility. The tests were divided into Phase A and Phase B. WCAP-13891, Revision 0, describes the Phase A tests; WCAP-14324 describes the Phase B tests. Phase A tests simulated ADS operation through the submerged sparger to evaluate the hydraulic performance of the sparger under various steamflow rates and to measure pressure pulses resulting from the discharge of steam into the quench tank simulating the IRWST. These results were used to define the dynamic forcing functions generated by the condensation of the steam. This information was used, in turn, to determine the dynamic loads imposed on the actual AP600 IRWST during sparger operation. Phase B tests developed functional requirements and assessed the performance of the ADS valves.

In response to RAI 220.001, Westinghouse stated that the AP600 ADS hydraulic tests were used to define loads on the AP1000 IRWST. Two tests were selected as representative of the sparger discharge pressures. One test simulated the pressure-time history corresponding to the ADS operating beyond 400 seconds after ADS initiation, when the RCS pressure is reduced and significant two-phase flow is discharged through the spargers. The other test simulated a pressure-time history representing the inadvertent opening of the second or third stage of the ADS at full pressure. The latter test is characterized by pure steamflow.

The response of the AP1000 IRWST to these time-history forcing functions is discussed in reference to RAI 220.009.

Westinghouse states that the time histories from these two tests are applicable to the AP1000 because the ADS valves and the ADS piping and spargers are identical for both the AP600 and the AP1000 designs. The valve opening times, flow areas, and fluid conditions are also the same. The ADS flow rate for the two-phase flow test is bounded by the value used for the AP1000 design. For the single-phase flow test, the important time is the initial time; the fluid conditions are similar.

Because the designs are identical and the fluid conditions for the tests used to determine the loads are bounding in one case and similar in the other, the staff finds the hydrodynamic loads on the IRWST for the AP1000 to be acceptable.

6.3 Passive Core Cooling System

The PXS is a safety-related system designed to perform the following safety-related functions:

- emergency core decay heat removal
- RCS emergency makeup and boration
- safety injection

- containment sump pH control

The PXS is located inside the containment, and consists of the following major subsystems and associated components:

- an IRWST
- a PRHR HX
- two CMTs
- an ADS
- two accumulators
- pH adjustment baskets
- associated piping, valves, instrumentation, and other related equipment

These PXS subsystems or components require only a one-time alignment of valves upon actuation. Once the initial actuation alignment is made, they rely solely on natural forces, such as gravity and stored energy, to operate. The use of active equipment or supporting systems, such as pumps, ac power sources, component cooling water, or service water, is not required.

DCD Tier 2, Figures 6.3-3 and 6.3-4 provide a general sketch of the PXS configuration. The IRWST is a large tank located above the elevation of the RCS loops that contains more than 2,234 m^3 (78,900 ft^3) of borated water. It is the source of low-pressure safety injection by gravity and the heat sink for the PRHR HX, which is submerged within it. The PRHR HX is connected to the RCS through an inlet line from one RCS hot-leg and an outlet line to the associated SG cold-leg plenum (RCP suction). The PRHR HX removes core decay heat by natural circulation. The CMTs, which are filled with borated water during normal operation, are located at an elevation above the RCS loops, and are connected to the RCS by pressure balance lines from the cold-legs, which maintain the CMTs at the RCS pressure. The outlet line from the bottom of each CMT provides an injection path to the DVI lines into the reactor. The ADS consists of four different stages of valves. The first three stages are connected to the top of the pressurizer and discharge through a sparger into the IRWST. The fourth-stage valves connect to the top of the RCS hot-legs and vent directly into the SG compartment. The ADS valves are actuated sequentially to depressurize the RCS to allow for gravity injection from the IRWST. The accumulators are filled with borated water that is pressurized with nitrogen gas and will inject, via the DVI lines, into the RCS when the RCS pressure falls below the accumulator pressure. The containment sump water pH control uses pH adjustment baskets containing granulated TSP, which dissolves when the containment sump water floodup reaches the baskets, to maintain the required recirculation sump pH during severe accident conditions.

The PXS is designed to mitigate design-basis events that involve a decrease in the RCS inventory such as a LOCA, or an increase or decrease in heat removal by the secondary system. For those non-LOCA events that result in an increase or decrease in heat removal by the secondary system, the PRHR HX and CMT are actuated by the protection and safety monitoring system (PMS) to remove core decay heat and provide makeup and boration for reactor coolant shrinkage. For events that reduce RCS inventory, the CMTs are actuated by the PMS to deliver borated water to the RCS via the DVI nozzles. As the CMTs drain down, the ADS valves are sequentially actuated to depressurize the RCS and establish the low-pressure conditions that allow injection from the accumulators, the IRWST, and the containment recirculation sump.

The staff's review of the PXS uses SRP Section 6.3 as guidance. Because the AP1000 PXS is quite different from the ECCS of the existing PWR designs, some SRP guidelines do not apply.

The staff reviewed the PXS for conformance with the following requirements:

- GDC 2, "Design Basis for Protection Against Natural Phenomena," as it relates to the seismic design of the SSCs the failure of which could cause an unacceptable reduction in the capability of the ECCS to perform its safety function

- GDC 4, as it relates to the dynamic effects associated with flow instabilities and loads

- GDC 5, "Sharing of Structures, Systems, and Components," as it relates to SSCs that are important to safety being prohibited from being shared among nuclear power units unless it can be demonstrated that sharing will not impair their ability to perform their safety function

- GDC 17, "Electric Power Systems," as it relates to the onsite and offsite electric power systems to permit functioning of the ECCS to provide sufficient capacity to ensure that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded and that the core is cooled during anticipated operational occurrences and accident conditions

- GDC 27, "Combined Reactivity Control Systems Capability," as it relates to the system being designed with the capability to ensure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained

- GDC 34, "Residual Heat Removal," as it relates to the ability of the residual heat removal system to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded

- GDC 35, "Emergency Core Cooling," GDC 36, "Inspection of Emergency Core Cooling System," and GDC 37, "Testing of Emergency Core Cooling System," as they relate to the ability of the ECCS to provide an abundance of core cooling to transfer heat from the core at a rate so that fuel and clad damage will not interfere with continued effective core cooling, to permit appropriate periodic inspection of important components, and to permit appropriate periodic pressure and functional testing

- 10 CFR 50.46 and Appendix K to 10 CFR Part 50, as they relate to analysis of the ECCS performance to ensure that it is accomplished in accordance with an acceptable evaluation model.

6.3.1 Design Bases

In DCD Tier 2, Section 6.3.1, Westinghouse describes the AP1000 PXS design bases. The PXS is designed to perform its safety-related functions on the basis of the following considerations:

- It has component redundancy to perform safety-related functions for postulated design-basis events.
- Components are designed and fabricated according to industry-standard quality groups commensurate with their intended safety-related functions following events such as fire, internal missiles, or pipe breaks.
- Components are tested and inspected at appropriate intervals, as defined by ASME Code, Section XI, and by TSs.
- Components are protected from the effects of external events, such as earthquakes, tornados, and floods.
- Components are sufficiently reliable, considering redundancy and diversity, to support the plant core melt frequency and significant release frequency goals.

The following sections describe the safety-related functional performance criteria of the PXS.

6.3.1.1 Emergency Core Decay Heat Removal

For non-LOCA events in which a loss of core decay heat removal capability via the SGs occurs, the PRHR HX is designed to automatically actuate to (1) remove core decay heat to prevent water relief through the pressurizer safety valves, (2) cool the RCS to 215.6 °C (420 °F) within 36 hours, with or without RCPs operating, (3) continue decay heat removal operation for an indefinite time in a closed-loop mode of operation in conjunction with the PCS, and (4) sufficiently reduce RCS temperature and pressure during an SG tube rupture (SGTR) event to terminate breakflow, without overfilling the SG.

6.3.1.2 RCS Emergency Makeup and Boration

For non-LOCA events that result in an inadvertent cooldown of the RCS, such as a steamline break, the PXS will automatically provide sufficient borated water to make up for reactor coolant shrinkage, counteract the reactivity increase caused by the system cooldown, allow for decay heat removal, prevent actuation of the ADS, and eventually bring the RCS to a subcritical condition.

6.3.1.3 Safety Injection

The PXS provides sufficient water to the RCS to mitigate the effects of a LOCA. In the event of a large-break LOCA, up to and including a cold-leg guillotine break, the PXS rapidly refills the reactor vessel, refloods the core, and continuously removes the core decay heat so that the performance criteria for ECCSs are satisfied.

The ADS valves are designed so that the PXS will satisfy the small-break LOCA performance requirements and provide effective long-term core cooling.

6.3.1.4 Safe Shutdown

Establishing a safe-shutdown condition requires maintenance of the reactor in a subcritical condition and adequate cooling to remove residual heat. One of the functional requirements for the PXS is that the plant be brought to a stable condition using the PRHR HX for non-LOCA events. Because of the functional limitations of the safety-related PRHR HX in passive plant designs, the Commission, in an SRM issued June 30, 1994, approved the position proposed in SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Designs.” This position accepts 215.6 °C (420 °F) or below, rather than the cold shutdown specified in RG 1.139, “Guidance for Residual Heat Removal,” as the safe stable condition that the passive decay heat removal system must be capable of achieving and maintaining following non-LOCA events. The PXS establishes safe shutdown by providing the necessary reactivity control to maintain the core in a subcritical condition, and by providing residual heat removal capability to maintain adequate core cooling. DCD Tier 2, Section 7.4, discusses the systems required for safe shutdown.

For non-LOCA events, the PRHR HX, in conjunction with the PCS, has the capability to bring the plant to a stable safe-shutdown condition, cooling the RCS to about 215.6 °C (420 °F) in 36 hours, with or without the RCPs operating.

The CMTs automatically provide emergency coolant makeup and boration to the RCS as the temperature decreases and pressurizer level decreases, opening the CMT injection valves upon a low-pressurizer level. The PXS can maintain stable plant conditions for an extended period of time in this mode of operation, depending on the reactor coolant leakage, without ADS actuation. For example, with reactor coolant leakage at the TS limit of 38 L/min (10 gpm), stable plant conditions can be maintained for at least 10 hours.

The ADS automatically actuates when the liquid volume in the CMTs decreases below the ADS actuation setpoints. The ADS valves are powered by the Class 1E dc batteries which provide power for at least 24 hours. A timer, which measures the time that ac power sources are unavailable and, therefore, the time the Class 1E batteries are being discharged, is used to automatically actuate the ADS if offsite and onsite ac power sources are lost for 24 hours. Therefore, for LOCAs or other postulated events in which ac power sources are lost, or when the CMT levels are sufficiently low, the ADS is automatically actuated. This results in injection from the accumulators, and subsequently from the IRWST, once the RCS is nearly depressurized. For these conditions, the RCS depressurizes to saturated conditions at about 121.1 °C (250 °F) within 24 hours. The PXS can maintain the plant in this safe-shutdown condition indefinitely.

6.3.1.5 Containment Sump pH Control

The pH adjustment baskets of the PXS are capable of maintaining the postaccident pH conditions in the recirculation water within a range of 7.0 to 9.5 after containment floodup to enhance radionuclide retention in the containment sump and prevent SCC of containment components during long-term containment floodup.

6.3.2 System Design

The AP1000 PXS is a seismic Category 1, safety-related system located inside the containment. Therefore, the PXS is designed for a single NPP, and is not shared between units, as required by GDC 5. GDC 17 requires that an onsite and offsite electric power system be provided to permit functioning of SSCs important to safety. The PXS relies on natural forces to perform its safety functions. It does not rely on any active system, except for one-time alignment of dc-powered valves upon actuation. Therefore, no safety-related onsite or offsite ac electric power is needed for PXS functions. The PXS is designed to provide adequate core cooling for design-basis events. Redundant onsite safety-related Class 1E dc and UPS system power sources are provided to ensure that the system safety functions can be accomplished under conditions when all ac power is lost, and assuming a single failure has occurred coincident with an event.

The PXS design comprises the six major subsystems or components that function together in various different combinations to perform safety-related functions. A description of the six major subsystems and components follows. DCD Tier 2, Figures 6.3-1 and 6.3-2 depict the piping and instrumentation drawings of the PXS. DCD Tier 2, Table 6.3-2, contains a summary of equipment parameters for the major components.

6.3.2.1 Core Makeup Tanks

The CMTs provide RCS makeup and boration during non-LOCA events when the normal makeup system is unavailable or insufficient. For LOCA events, the CMTs provide high-pressure safety injection to the RCS.

The two CMTs are vertical, cylindrical tanks with hemispherical upper and lower heads located inside containment on the 32.6 m (107-ft) floor elevation, slightly above the RCS loops (the bottom inside surface of each CMT is at least 2.3 m (7.5 ft) above the DVI nozzle centerline). Each CMT, having a volume of 70.8 m^3 (2500 ft^3), is connected to the RCS through an inlet pressure balance line connecting to a cold-leg and a discharge line connected to a DVI line. Each CMT has an inlet diffuser, which is designed to reduce steam velocities entering the CMT during relatively large-size, small-break LOCAs, thereby minimizing potential water hammer. The CMTs are made of carbon steel, clad on the internal surfaces with stainless steel.

During normal operation, the CMTs are completely filled with cold, borated water of about 3400 ppm, and are maintained at the RCS pressure by the pressure balance line, which prevents water hammer upon initiation of the CMT injection. The inlet pressure balance line contains a normally open MOV, and is sized to supply sufficient steam to allow CMT injection for LOCAs, where the cold-leg becomes voided and higher CMT injection flows are required. The pressure balance line also includes a high point vent line, which has two manual isolation valves in series and discharges to the reactor coolant drain tank. The operator can open the isolation valves to remove and prevent the accumulation of noncondensible gases that could interfere with CMT operation. The discharge line has two parallel, normally closed, air-operated isolation valves that will open upon a loss of air pressure or electric power, or on control signal actuation, to begin CMT injection. Downstream of the AOVs, the outlet lines combine into one line, which contains two tilt-disc check valves in series to prevent backflow from the DVI line. The discharge line from each CMT contains a flow-tuning orifice to provide for field adjustment of the injection line resistance to establish the required flow rates for the associated plant conditions assumed in the CMT design. The flow-tuning orifice will be adjusted as part of the preoperational test program.

The CMT is actuated by the opening of the two parallel isolation valves in the discharge lines. There are two operating processes for the CMTs, water recirculation and steam-compensated injection. During water recirculation, hot water from the cold-leg enters the CMT, and the cold water in the tank is discharged to the RCS. This results in RCS boration and a net increase in the RCS mass. During the steam-compensated injection, steam is supplied through the cold-leg balance line to the CMT to displace the water that is injected into the RCS.

DCD Tier 2, Section 7.3.1.2.3 and Table 7.3-1 describe the actuation signals and logic, as well as the permissives and interlocks, to align the CMT for injection; Table 3.3.2-1 of the AP1000 TS specifies the actuation setpoints. DCD Tier 2, Table 15.0-4b, provides the discharge valve opening delay times used in the safety analyses.

6.3.2.2 Accumulators

The two accumulators are spherical tanks located on the containment floor just below the CMTs. The accumulators, each having a volume of 56.63 m^3 (2000 ft^3), are filled with borated water at a concentration of about 2600 ppm and pressurized with nitrogen gas to a pressure between 4.49 and 5.4 MPa (651 and 783 psia). Each accumulator is connected to one of the DVI lines. Each injection line contains an MOV, a flow-tuning orifice, and two swing-disc check valves in series. The MOV is normally open with power removed and locked out to prevent inadvertent isolation. The flow-tuning orifice provides for field adjustment of the injection line resistance. During normal operation, the accumulator is isolated from the RCS by the check valves. The accumulators have gas relief valves to protect them from overpressurization caused by leakage from the RCS. The system also includes the capability to remotely vent gas from the accumulator, if required. During a LOCA, when the RCS pressure falls below the accumulator pressure, the check valves open and the borated water is forced into the RCS by the gas pressure. The AP1000 accumulator check-valve application is identical to that for current plants.

6.3.2.3 In-Containment Refueling Water Storage Tank

The IRWST is a large, stainless-steel lined tank containing 2,234 m^3 (78,900 ft^3) of borated water with a boron concentration of about 2,600 ppm. The IRWST is a safety injection source, and also serves as the heat sink for the PRHR HX, which is submerged within it. The IRWST is connected to the RCS through both DVI lines. The IRWST is AP1000 Class C equipment, designed to meet seismic Category I requirements, and is constructed as an integral part of the containment internal structures. Its bottom is above the RCS loop elevation (the bottom inside surface is at least 1.04 m (3.4 ft) above the DVI nozzle centerline) so that the borated refueling water can drain and inject by gravity into the RCS after the RCS is depressurized. Each injection line from the IRWST contains an MOV, which is normally open with power removed and locked out. The injection line contains two parallel lines, each with a check valve and a squib valve in series. RCS injection from the IRWST is possible only after the RCS has been depressurized by the ADS or a LOCA. Squib valves in the IRWST injection lines open automatically on a fourth-stage ADS initiation signal. Check valves open when the reactor pressure decreases below the IRWST injection head.

After the accumulators, CMTs, and IRWST inject, the containment is flooded to a level sufficient to provide recirculation flow through the gravity injection lines back into the RCS. There are two containment recirculation lines from the containment sump, each connecting to an IRWST injection line. Each recirculation line contains two parallel lines, one having a normally open MOV and a squib valve in series, and the other having a check valve and a squib valve in series. When the IRWST level decreases to a low level, the recirculation line squib valves automatically open to provide redundant flowpaths from the containment to the reactor.

DCD Tier 2, Section 7.3.1.2.2 and Table 7.3-1 describe the actuation signals and logic, as well as the permissives and interlocks, to align the IRWST injection and containment recirculation; Table 3.3.2-1 of the AP1000 TS specifies the actuation setpoints.

The IRWST and the containment recirculation sump are each provided with two separate screens to prevent debris from entering the reactor and blocking core cooling passages during a LOCA. These screens are oriented vertically, and located at the bottom of the opposite ends of the IRWST and the containment sump along the walls about 0.6 m (2 ft) above the floor. They are designed to comply with RG 1.82, “Water Sources for Long-Term Recirculation Cooling Following a Loss-of-Coolant Accident.” The IRWST is lined with stainless steel and does not contain material either in the tank or the recirculation path that could plug the outlet screens. The TS require visual inspections of the screens during every refueling outage to ensure they are not restricted by the debris. DCD Tier 2, Section 6.3.2.2.7, discusses the design of the IRWST and recirculation screens, as well as the design criteria. Section 6.2.1.8 of this report discusses the staff’s evaluation of the IRWST and recirculation screens.

6.3.2.4 pH Adjustment Baskets

The PXS utilizes pH adjustment baskets to control the postaccident pH level in the containment sump within a range of 7.0 to 9.5. The baskets, which contain at least 12,518 kg (27,540 lbs) of granulated TSP, have a mesh front and are located below the minimum postaccident flood-up level so that chemical addition is initiated passively when the sump water reaches the baskets. The baskets are placed at least 0.3 m (1 ft) above the floor (the pH baskets are located below plant Elevation 32.7 m (107'-2") to reduce the chance that water spills in containment will dissolve the TSP.

The baskets are made of stainless steel with a mesh front that readily permits contact with water. Section 15.3 of this report evaluates the adequacy of the pH adjustment baskets.

6.3.2.5 Passive Residual Heat Removal Heat Exchanger

The PRHR HX consists of inlet and outlet channel heads connected by 689 vertical C-shaped tubes, 1.9 cm (0.75 in) in diameter. The tubes are supported and submerged inside the IRWST with the top of the tubes several feet below the IRWST water surface. The IRWST acts as a heat sink for the HX. The design heat transfer rate and flow are 2.11E+11 J/hr (2.00E+8 BTU/hr) and 2.28E+5 kg/hr (5.03E+5 lb/hr), respectively, as specified in DCD Tier 2, Table 6.3-2. The PRHR HX is connected to the RCS by an inlet line from one hot-leg (through a tee from one of the fourth-stage ADS lines) and an outlet line to the associated SG cold-leg plenum (RCP suction).

The PRHR HX performs emergency core decay heat removal for events not involving a loss of coolant. The HX is elevated above the RCS loops to induce natural circulation flow through the PRHR HX when the RCPs are not available. The PRHR HX inlet line contains a normally open MOV. This alignment maintains the HX full of reactor coolant at the RCS pressure. The outlet line contains two parallel, normally closed, AOVs that open upon loss of air pressure or on control signal actuation, and a normally open, manually operated valve in series. The two parallel valves in the discharge line ensure an available flowpath for the single-failure assumption of an inoperable valve in the safety analysis. DCD Tier 2, Table 15.0-4b, provides the discharge valve opening time delays assumed in the safety analyses. The water temperature in the HX is about the same as the water temperature in the IRWST, so that a thermal driving head is established and maintained during plant operation. The PRHR HX piping arrangement also allows for actuation of the HX with the RCPs operating, which provide forced flow in the same direction as the natural circulation. If the pumps are operating and subsequently trip, natural circulation continues to provide the driving force for HX flow. The PRHR HX flow and inlet and outlet temperatures are monitored by indicators and alarms. The operator can take action, as required, to meet the TS requirements or follow emergency operating procedures for control of the PRHR HX operation.

The PRHR HX has a high point vent, which is a vertical pipe stub on the top of the inlet piping high point that serves as a gas collection chamber. Level detectors indicate when the gases have collected in this area. The operator can open manual valves to locally vent these gases to the IRWST.

The PRHR HX, in conjunction with the PCS, can provide core cooling for an indefinite period of time. The operation of the PRHR HX results in the steaming of the IRWST water. Steam condensation occurs on the steel containment vessel, and the condensate returns to the IRWST through a safety-related gutter arrangement located at the operating deck level. The gutter normally drains to the containment sump, but will direct the gutter overflow to the IRWST when safety-related isolation valves in the gutter drainline shut at the initiation of the PRHR. Recovery of the condensate maintains the PRHR HX heat sink for an indefinite period of time.

DCD Tier 2, Section 7.3.1.2.7 and Table 7.3-1 describe the actuation signals and logic, as well as the permissives and interlocks, to align the PRHR HX for heat removal; Table 3.3.2-1 of the AP1000 TS specifies the actuation setpoints. DCD Tier 2, Table 15.0-4b, provides the discharge valve opening delay times used in the safety analyses.

6.3.2.6 Automatic Depressurization System

The ADS has a total of 20 valves divided into 2 identical groups, each consisting of 4 different stages of valves. Each of the first three stages has two normally closed, dc MOVs in series, one termed an isolation valve and the other a control or depressurization valve. The isolation valves are gate valves, and the control valves are globe valves. The fourth stage in each group has a common header connected directly to the top of an RCS hot-leg. The header branches into two lines, each containing a normally open motor-operated gate valve and a squib valve in series. The fourth-stage valves vent directly to the SG compartment. DCD Tier 2, Section 5.4.6.2, specifies that the first-stage ADS valves are motor-operated, 10-cm (4-in.) valves, the second- and third-stage valves are 20-cm (8-in.) valves, and the fourth-stage valves are 35.6-cm (14-in.) valves.

The first three stages in each group have a common inlet header connected to the top of the pressurizer. The outlets of each group of the first three stages are combined into a common discharge line to a sparger. The sparger has four branch arms inclined downward. The sparger midarms are submerged below the normal water level in the IRWST and are designed to distribute steam into the IRWST, thereby promoting more effective steam condensation. The installation of the spargers prevents undesirable and excessive dynamic loads on the IRWST. Each sparger is sized to discharge at a flow rate that supports the ADS performance to depressurize the RCS to allow adequate PXS injection. The common discharge line also has a vacuum breaker to help prevent water hammer following ADS operation by limiting the pressure reduction caused by steam condensation in the discharge line, and thus limiting the potential for liquid backflow from the IRWST.

The ADS valves are designed to automatically open when their actuation setpoints are reached, and remain open for the duration of an automatic depressurization event. The Stages 1, 2, and 3 ADS valves open sequentially. The isolation valves in each stage open first, followed by the control valves, which are designed to open relatively slowly, after a short time delay. DCD Tier 2, Section 7.3.1.2.4, discusses the ADS actuation logic and Table 7.3-1 summarizes this information. The first stage valves automatically actuate on the CMT Low-1 level signal; the second- and third-stage valves actuate subsequently with preset time delays between stages. The fourth-stage valve actuates upon the coincidence of a CMT Low-2 level and low RCS pressure, following a preset time delay after the third-stage depressurization valves are opened. The fourth-stage valves can also be opened upon the occurrence of coincidence loop 1 and loop 2 hot-leg levels below the Low-2 setpoint for a duration exceeding a time delay. This signal is automatically blocked when the pressurizer water level is above the P-12 setpoint to reduce the possibility of a spurious signal. DCD Tier 2, Table 15.6.5-10, provides a list of ADS parameters, including the CMT levels when the various ADS stage valves actuate, the actuation delay times, minimum valve flow areas, and valve opening times. The operators can also manually open the first-stage valves to a partially open position to perform a controlled RCS depressurization. The operator can also manually initiate the fourth-stage valves. The manual initiation signal is interlocked to prevent actuation until either the RCS pressure has decreased below a preset setpoint, or until the signals that control the opening sequence of the first three stages have been generated.

6.3.2.7 <u>Low-Differential, Pressure Opening Check Valves</u>

Passive core cooling systems contain several check valves designed to operate with low-differential pressures which could affect the passive system reliability. Section B, "Definition of Passive Failure," of SECY-94-084, describes a Commission-approved position (SRM issued June 30, 1994) to maintain current licensing practices for passive component failures in passive LWR designs. The position also redefines check valves (except for those whose proper function can be demonstrated and documented) in the passive safety systems as active components subject to single-failure consideration.

The AP1000 PXS has been specifically designed to treat check valve failures-to-reposition as active failures. It assumes that normally closed check valves fail to open and normally open check valves fail to close. Check valves that remain in the same position before and after an event are not considered active failures. Exceptions to this treatment in the PXS are made for the accumulator and CMT check valves. The treatment of the accumulator check valves is consistent with the treatment of these specific check valves in currently licensed plant designs because the accumulator pressure will eventually create a large pressure differential to force open the valves as the RCS pressure falls. The CMT check valve exception to active failure treatment is discussed below.

DCD Tier 2, Section 1.9.5.3.2, states that the AP1000 is designed with redundancy for the check valve applications in the CMT discharge lines, the IRWST gravity injection lines, and the containment isolation lines that use check valves. The redundancy and diversity in the design among these multiple safety-related flowpaths is sufficient to accommodate the single failure of a check valve to reposition as required to perform its safeguard function. The staff agreed with Westinghouse's position, and used this position to evaluate the appropriateness of the check valve arrangements in the PXS as described below.

Both the IRWST and the containment recirculation injection lines contain normally closed, simple swing check valves, which must change position to perform their safety functions. Therefore, these check valves are considered active components subject to the single-failure assumption. Each IRWST injection line contains two parallel paths, each having a check valve and a squib valve in series. The redundant parallel paths design assures operability of the IRWST injection with a single failure of a check valve. The containment recirculation injection line also contains two redundant parallel paths, one having a check valve and a squib valve in series, and the other having a normally open MOV and a squib valve in series.

Each CMT injection line contains two tilt-disc check valves in series to prevent backflow from the DVI line. However, these tilt-disc check valves are biased open during normal plant operation and do not have to change position to perform their safety function to open the CMT injection lines. Only a low probability exists that these check valves will not reopen within a few minutes after they have cycled closed during accumulator operation. Therefore, they are considered passive components, not subject to single-active-failure consideration for the opening function. However, a single-active-failure has been taken into account for the closing function of these check valves by providing two check valves in series.

Each accumulator injection line contains two normally closed, swing check valves in series to prevent the RCS backflow. However, these check valves are similar to the check valves used in current PWR applications and are in the closed position with a differential pressure of about 10.6 MPa (1550 psid) during normal operation. They are not subject to the degradation from flow operation or impact loads caused by sudden flow reversal and seating. During a LOCA, these check valves will be forced open by a large differential pressure created by the accumulator pressure as the RCS depressurizes. Therefore, as stated above, they are not subject to single-active-failure consideration.

The staff finds that the check valve arrangements in the PXS are designed with redundancy to accommodate the single active failure of a check valve to reposition as required to perform its safeguard function, and are therefore acceptable.

6.3.2.8 System Reliability

The AP1000 PXS is designed to satisfy a variety of requirements to ensure its availability and the reliability of its safety functions, including redundancy (e.g., for components, power supplies, actuation signals, and instrumentation), equipment testing to confirm operability, procurement of qualified components, and provisions for periodic maintenance. In addition, the design provides protection against single active and passive component failures; spurious failures; physical damage from fires, flooding, missiles, pipe whip, and accident loads; and environmental conditions, such as high-temperature steam and containment floodup. These requirements are specified in GDC 2, 4, 34, 35, 36, and 37.

To ensure system operability and allow for immediate corrective actions, the PXS equipment conditions are monitored with indications and/or alarms in the MCR to alert the operator of equipment conditions outside of the TS limits. The monitored parameters include the CMT level, temperature, and inlet line noncondensible gas volume; accumulator level and pressure; IRWST level and temperature; and PRHR HX inlet line noncondensible gas volume.

6.3.2.8.1 Redundancy and Single-Failure Consideration

The PXS system is designed with sufficient redundancy to withstand credible single active and passive failures. The AP1000 has been specifically designed to treat check valve failures-to-reposition as active failures. Check valves that remain in the same position before and after an event are not considered active failure. As discussed in Section 6.3.2.7 of this report, the accumulator check valve opening and the CMT check valve reopening are the two exceptions. Chapter 15 of the DBA analyses considers single active failures. In addition, for those valves that reposition to initiate safety-related system functions, the valve reposition times are less than the times assumed in the accident analyses.

A passive failure in a fluid system is a breach in the fluid pressure boundary or mechanical failure that adversely affects a flowpath. SECY-94-084 states the Commission-approved position that, consistent with current licensing practices, passive advanced light-water reactor (ALWR) designs need not assume passive component failures in addition to the initiating failure in the application of single-failure criterion to assure safety of the nuclear power plant. In addition, the staff only considers, on a long-term basis, passive component failures in fluid systems as potential accident initiators, in addition to initiating events. The AP1000 PXS can sustain a single passive failure during the long-term cooling phase and still retain an intact flowpath to the core to supply sufficient flow to keep the core covered and to remove decay heat. The PXS flowpaths are separated into redundant lines, either of which can provide minimum core cooling functions and return spill water from the floor of the containment back to the RCS. For the long-term PXS function, adequate core cooling capacity exists with one of the two redundant flowpaths.

The staff reviewed the piping diagrams of DCD Tier 2, Figures 6.3-1 through 6.3-4, to evaluate the functional reliability of the system in the event of single failures. The existence of the redundancy required by the single active failure is confirmed.

DCD Tier 2, Table 6.3-3, provides a summary of the failure mode and effect analysis of the PXS active components. To determine the effect on system operation, each of the valves in the PXS (including check valves, isolation valves, AOVs or MOVs, and squib valves) and the Class 1E dc and UPS system distribution switchgear division were examined for failure modes, as well as failure detection methods, for all design-basis events to determine the effect on system operation.

6.3.2.8.2 Valve-Opening Lag Times

For those valves that reposition to initiate safety-related system functions, the valve repositioning times are less than the times assumed in the accident analyses, as specified in DCD Tier 2, Table 15.0-4b. These lag times refer to the time after initiation of the safeguards actuation signal.

6.3.2.8.3 Potential Boron Precipitation

Boron precipitation in the reactor vessel is prevented by sufficient flow of PXS water through the core to limit the increase in boron concentration of the water remaining in the reactor vessel. Water, along with steam, leaves the core and exits the RCS through the fourth-stage ADS lines. The results of long-term cooling analysis of various breaks, presented in DCD Tier 2, Section 15.6.5.4C.4, indicate that venting of core 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 this report presents the staff's evaluation of this issue.

6.3.2.8.4 Testing and Inspection

The AP1000 PXS systems and components are designed to permit periodic inspection and testing of the operability of the system throughout the life of the plant, as required by GDC 36, "Inspection of Emergency Core Cooling System," and GDC 37. DCD Tier 2, Section 6.3.6, describes the inspection and testing requirements, including the preoperational and inservice inspection and testing. Preoperational inspections are performed to verify that important elevations associated with the PXS components are consistent with the accident analyses. DCD Tier 2, Section 14.2.9.1.3 describes the preoperational testing of the PXS. This testing includes valve inspection and testing, flow testing, and verification of heat removal capability.

Two basic types of inservice testing are performed on the PXS components, including periodic exercise testing of active components during power operation, and operability testing of specific PXS features during plant shutdown. To support inservice test performance, the PXS includes (1) remotely operated valves that can be exercised during routine plant maintenance, (2) level, pressure, flow, and valve position instrumentation to monitor required PXS equipment during plant operation and testing, and (3) permanently installed test lines and connections for operability testing. DCD Tier 2, Section 3.9.6.2, provides a description of the inservice testing of valves. DCD Tier 2, Tables 3.9-16 and 3.9-17, respectively, specify the valve inservice test requirements and system level operability test requirements.

6.3.2.8.5 Seismic and Equipment Classifications

The AP1000 PXS is a safety-related system, and all the subsystems are designed to meet seismic Category 1 requirements, as defined by RG 1.29. DCD Tier 2, Table 3.2-3, specifies the seismic category and the quality group classification of various system components. The PXS components are designed to meet the requirements of seismic Category 1 SSCs and withstand the effects of an SSE and remain functional. Because all the PXS subsystems rely on natural forces, such as gravity and stored energy, to perform their safety functions, they require no supporting systems, the failure of which could have an adverse effect on the PXS. No failure of a non-safety-related system could reduce the functioning of the PXS. Therefore, the PXS meets position C.2 of RG 1.29 and therefore fulfills GDC 2 requirements.

Portions of the PXS, such as the PRHR HX, CMT, and ADS, which are also part of the RCPB are designated AP1000 Class A components. For the portions of the PXS that are not part of the RCPB, RG 1.26 recommends that the ECCS systems be classified as Quality Group B. DCD Tier 2, Table 3.2-3, lists many PXS components as AP1000 Class C components. These Class C components include the following:

- the accumulators and injection line piping system up to the check valves
- the IRWST injection and containment recirculation piping up to the injection line check valves
- ADS Stages 1, 2, and 3 discharge spargers
- pH adjustment baskets

However, as discussed in Section 3.2.2 of this report, the staff determined that AP1000 Class C categorization for these portions of the PXS is acceptable. This finding is based on its evaluation of the design bases provided by Westinghouse, as well as the commitment stated in DCD Tier 2, Section 3.2.2.5 that for systems that provide ECC functions, full radiography, in accordance with the requirements of ASME Code, Section III, Subarticle ND-5222, will be conducted on the piping butt welds during construction.

6.3.2.8.6 Valves

Manual valves are generally used as maintenance isolation valves. When used for this function, they are under administrative controls. They are located so that no single valve can isolate redundant PXS equipment, or they are provided with position indication and alarms in the MCR to indicate mispositioning.

DCD Tier 2, Table 6.3-1, provides a list of the remotely actuated valves in the PXS subsystems, as well as their normal positions, actuated positions, and failed positions. These valves have their controls and valve position indication in the control room. The AP1000 TS requires that remotely operated isolation valves, such as the isolation valves on the PRHR inlet line, the CMT cold-leg balance lines, the accumulator and IRWST discharge lines, and the ADS fourth-stage MOVs, which are normally open and remain open during PXS operation, be verified fully open every 12 hours during normal plant operation. These isolation valves also have interlock features to ensure they are open for the PXS operation. DCD Tier 2, Section 7.6.2, “Availability of Engineered Safety Features,” discusses the interlock features, and Section 7.6 of this report discusses the staff’s evaluation of these features. These isolation valves do not receive safeguards actuation signals. They are normally manually controlled, but are also controlled by actuation control circuits, which have a function to direct the valve to open upon receipt of a “confirmatory open” signal in case the valves are closed. The use of confirmatory open signals to open these isolation valves, which are provided by the safeguards signals to actuate the respective PXS subsystem, provides a means to automatically override bypass features that are provided to allow these isolation valves to be closed for short periods of time. The accumulator and IRWST injection isolation valves have interlocks, and have their control power locked out during normal plant operation, in accordance with BTP Instrumentation and Control System Branch (ICSH)-18, to prevent their inadvertent operation.

The check valves in the IRWST injection line, the containment recirculation lines, the accumulator discharge lines, and the CMT injection lines have nonintrusive position indications and alarms in the MCR to alert the operators to valve mispositioning.

Explosively opening squib valves are used to isolate the IRWST injection line, the containment recirculation lines, and the ADS Stage 4 valves. These squib valves are used to provide zero leakage during normal operation, and to provide reliable opening during an accident. After they are open, they are not required to reclose. These valves are arranged in series with another valve. A valve open position sensor is provided for these valves.

6.3.2.8.7 Instrumentation

The AP1000 PXS design is provided with instrumentation for monitoring PXS components during normal plant operation and postaccident operation with indications and alarms in the MCR. The PRHR HX has pressure and inlet temperature indications to detect reactor coolant leakage into the PRHR HX. The PRHR HX also has two flow channels to monitor and control PRHR HX operation. Each accumulator has two pressure and two level channels to confirm that the pressure and level are within the bounds of the safety analysis assumptions. The IRWST has four temperature and four level channels to monitor the temperature and level. Each CMT has temperature indications in the inlet and outlet lines to determine if there is sufficient thermal gradient for system operation, and to detect RCS leakage into the CMT through the DVI line, respectively. Each CMT also has a level instrument, as discussed below, to be used for control of ADS actuation. Each DVI line has temperature indication to detect RCS leakage through the DVI line to the CMT, accumulator, or the IRWST. The containment has three level channels and four radiation channels. DCD Tier 2, Chapter 7, discusses the AP1000 instrumentation and controls, and Chapter 7 of this report discusses the staff's review of these components.

DCD Tier 2, Section 6.3.7.4.1, provides a design description of the CMT-level instrumentation using differential pressure instruments. DCD Tier 2, Figure 6.3-1, depicts the arrangement of the CMT differential pressure level instrument. Each CMT has 10 level channels. Two wide-range level channels, which are not qualified for postaccident monitoring, are used to confirm that the CMT is maintained at full water level during normal operation. Two sets of four narrow-range level channels, which are qualified for postaccident monitoring, are used for actuation of the ADS Stage 1 and Stage 4 valves. As discussed in Section 7.3 of this report, the staff found the CMT-level instrumentation to be acceptable.

6.3.2.8.8 Protection Provisions

The AP1000 PXS design incorporates specific design features that preclude water hammer and excessive dynamic loads, as required by GDC 4. These design features include the installation of the ADS spargers in the IRWST, the CMT inlet diffuser, sloping lines, and maintaining pressure in standby components. Various sections in the DCD describe measures taken to protect the system from damage that might result from various events. DCD Tier 2, Section 3.6, discusses protection against dynamic effects associated with piping rupture. DCD Tier 2, Section 3.9.3, discusses the load combinations, stress limits, and analytical methods for structural evaluation of the PXS for various plant conditions; DCD Tier 2, Section 3.9.2, discusses the requirements for dynamic testing and analysis. DCD Tier 2, Sections 3.7, 3.8, and 3.10 discuss seismic design. DCD Tier 2, Section 3.1.1, discusses environmental qualification of equipment. DCD Tier 2, Sections 3.5 and 9.5.1, respectively, discuss protection against missiles and from fire. The staff's evaluations of these DCD sections are discussed in the corresponding sections of this report.

6.3.3 Performance Evaluation

The AP1000 PXS is designed to mitigate design-basis events that involve a decrease in RCS inventory, an increase or decrease in heat removal by the secondary system, or events that can occur during shutdown operation.

6.3.3.1 <u>Shutdown Events</u>

During plant shutdown conditions, some of the PXS equipment is isolated to allow for maintenance of the system, and the RNS may not be available because it is not a safety-related system. As a result, gravity injection is automatically actuated when required to provide core cooling during shutdown conditions before refueling cavity floodup. In addition, the operator can manually actuate other PXS equipment, such as the PRHR HX, to provide core cooling during shutdown conditions if the equipment does not automatically actuate. Events that occur during shutdown conditions are characterized by slow plant responses and mild thermal-hydraulic transients. DCD Tier 2, Section 6.3.3.4, provides an evaluation of the PXS capability to mitigate the following four shutdown events:

(1) loss of startup feedwater during hot standby, cooldowns, and heatups
(2) loss of RNS cooling with the RCS pressure boundary intact
(3) loss of RNS cooling during midloop operation
(4) loss of RNS cooling during refueling

In DCD Tier 2, Section 19E.4, the applicant provided a more complete shutdown evaluation of applicable design-basis transients and accidents postulated to occur during shutdown operations. For each event category discussed in DCD Tier 2, Chapter 15, the applicant identified the limiting case and evaluated, for shutdown operations, the effects of plant control parameters, neutronic and thermal-hydraulic parameters, and ESFs on plant transient responses, such as departure from nucleate boiling ratio, peak RCS pressure, and peak cladding temperature. Section 19.3 of this report presents the staff's evaluation of shutdown operation. The staff concludes that the PXS with the shutdown configurations (to allow for maintenance of the system) is capable of coping with all events initiated during shutdown operation. Therefore, it is acceptable.

6.3.3.2 Power Operation Events

For non-LOCA events initiated during power operation, the PRHR HX is actuated by the PMS to remove core decay heat when any of the actuation conditions (e.g., SG Low wide-range level, SG low narrow-range level coincident with startup feedwater low flow, or CMT actuation) is reached. For LOCAs, the primary protection is provided by the CMTs and accumulators. When any of the PXS actuation conditions (e.g., low-pressurizer pressure or level, low-steamline pressure, high-containment pressure, or low SG level coincident with high RCS hot-leg temperature) is reached, the PMS will actuate the CMTs to deliver borated water to the RCS by means of the DVI nozzles. As the CMTs drain down, the ADS valves are sequentially actuated to depressurize and establish RCS pressure conditions that allow injection from the accumulators, and then from the IRWST and the containment recirculation sump. The accumulators deliver flow to the DVI line whenever RCS pressure drops below the tank static pressure. The IRWST provides gravity injection once the RCS pressure is reduced below the injection head from the IRWST. The PXS flow rates vary depending upon the type of event and its characteristic pressure transient. Therefore, an injection source is continuously available. In addition to initiating PXS operation, the PXS actuation conditions also initiate other automatic-action safeguards, including reactor trip, RCS pump trip, feedwater isolation, and containment isolation.

DCD Tier 2, Chapter 15, provides an evaluation of the design-basis events, and DCD Tier 2, Section 6.3.3, provides a summary of events that result in the actuation of the PXS to demonstrate functional performance capability of the PXS. An inadvertent opening of an SG relief or safety valve and a steam system pipe failure are among the non-LOCA events that result in an increase in heat removal by the secondary system. A loss of main feedwater and a feedwater system pipe failure are among the events that result in a decrease in heat removal by the secondary system. A single SGTR, LOCAs, and a complete severance of a single PRHR

HX tube are among the events that could result in a decrease in RCS inventory. DCD Tier 2, Sections 15.1, 15.2, and 15.6, respectively, analyzed these events. DCD Tier 2, Section 15.6, did not analyze a postulated double-ended rupture of one PRHR HX tube. The total area of a double-ended rupture of the PRHR HX is less than a 2.5-cm- (1-in.-) equivalent diameter break. With one tube ruptured, the PRHR HX remains essentially unaffected in terms of its heat removal capability. The PRHR tube rupture is nonlimiting and is covered by the effect of postulating a hot-leg or cold-leg break location considered in the break spectrum. DCD Tier 2, Section 15.6.5.4C, analyzes the post-LOCA, long-term cooling.

Chapter 15 of this report discusses the evaluation of the safety analyses of the design-basis events. In general, the design-basis analyses take credit for safety-related systems and components for mitigation of events. Consideration is given to operation of non-safety-related systems that could affect the event results. Section 15.1.2 of this report addresses the non-safety-related systems assumed in the design-basis analyses. A non-safety-related system or component is assumed to be operational when (1) its operation has an adverse effect that results in a more limiting transient, (2) a detectable and nonconsequential random, independent failure had to occur in order to disable the system, and (3) it is used as backup protection. Though GDC 17 regarding the requirements of onsite and offsite power supplies does not apply to the PXS, the effects of a loss of offsite power on the RCP trip and the results of transients and accidents are considered in the design-basis safety analysis. This complies with GDC 17. In addition, the analyses of the postulated accidents assume that the most reactive control rod stuck out of the core complies with GDC 27. The staff found the Chapter 15 design-basis analyses and the assumptions of the operation of non-safety-related systems and components, as well as other single-failure assumptions, to be acceptable.

The results of the Chapter 15 analyses demonstrate the appropriateness of the PXS performance for mitigation of the design-basis events. This complies with (1) GDC 34, in that the PRHR system is capable of transferring the decay heat and other residual heat from the core, such that the specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded, (2) GDC 35, in that the PXS provides abundant emergency core cooling capability following LOCAs so that fuel and cladding damage that could interfere with continued effective core cooling is prevented, and clad metal-water reaction is limited to a negligible amount, and (3) 10 CFR 50.46, in that the ECCS cooling performance is calculated in accordance with an acceptable evaluation model for the postulated LOCA break spectrum to demonstrate that the acceptance criteria specified in 10 CFR 50.46(b) are met.

The computer programs used for the analyses of these design-basis events are, respectively, LOFTRAN for the non-LOCA events, LOFTTR2 for the single SGTR event, NOTRUMP for small-break LOCAs, and WCOBRA/TRAC for large-break LOCAs and long-term cooling. Chapter 21 of this report discusses the review of these codes, as well as the test programs.

6.3.4 Post-72 Hour Actions

The AP1000 design relies on passive safety-related systems and equipment to automatically establish and maintain safe-shutdown conditions for the plant following design-basis events, assuming the most limiting single failure. These passive safety systems are designed with sufficient capability to maintain safe-shutdown conditions for 72 hours, without operator actions and without non-safety-related onsite or offsite power. Only one potential need exists for the containment inventory makeup to provide long-term core cooling because of containment leakage.

For the AP1000 PXS, the IRWST serves as the heat sink for the PRHR HX. During extended PRHR HX operation, steam from the IRWST is condensed by the PCS and the condensate returns to the IRWST by means of the safety-related gutter. This closed loop operation can continue indefinitely provided that no leakage through the containment occurs. For long-term core cooling, however, there is a potential need for operator action to provide containment inventory makeup, which is directly related to the leak rate from the containment. DCD Tier 2, Section 6.3.4, states that, with the maximum allowable containment leak rate, makeup to the containment is not needed for about 1 month. The AP1000 RNS design is equipped with a safety-related connection to align a temporary makeup source to containment. Therefore, the long-term cooling capability of the PXS is assured.

DCD Tier 2, Section 1.9.5.4, and WCAP-15985, Revision 2, "AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process," dated August 2003, describe the AP1000 design for the post-72 hour support actions required following an extended loss of these non-safety-related systems for the safety-related functions. The AP1000 design includes both onsite equipment and safety-related connections for use with transportable equipment and supplies to provide certain extended support actions. With regard to the PCS, the support actions use one of the two PCS recirculation pumps powered by an ancillary diesel generator or a portable, engine-driven pump that connect to a safety-related makeup connection that provides makeup water to the PCS water storage tank to maintain external containment cooling waterflow, and therefore provide the containment cooling and ultimate heat sink. Section 22.5.6 of this report describes the staff's evaluation of this post-72-hour support action. The staff concluded that since all equipment required for post-72 hour actions is onsite and consumable supplies are sufficient to last 7 days, the post-72 hour actions for the AP1000 are acceptable.

6.3.5 Limits on System Parameters

The plant TSs establish PXS operability requirements for reactor operation. TS 3.4.12 through 3.4.14, and 3.5.1 through 3.5.8 specify the limiting conditions for operation and SR of various PXS subsystems. In addition, planned maintenance on the PXS equipment is accomplished during refueling operations when the core temperatures and decay heat levels are low, and the IRWST water is in the refueling cavity. The TSs also provide the principal system parameters, the number of components that may be out of operation during testing, and the allowable time for operation in a degraded status. Chapter 16 of this report addresses the staff's evaluation of the TSs.

6.3.6 Conclusions

The staff reviewed DCD Tier 2, Section 6.3, and other relevant material regarding the AP1000 PXS design, including piping and instrumentation diagrams, failure modes and effects analyses, and the design specifications for essential components. The staff reviewed the AP1000 design bases and design criteria for the PXS, as well as the manner in which the design conforms to these criteria and bases. The staff concludes that the AP1000 PXS design meets the guidelines of SRP Section 6.3 and the requirements of the following GDC:

- GDC 2—The PXS is designed to meet the seismic Category 1 requirements and remain functional following an SSE.

- GDC 4—The PXS design incorporates features that preclude water hammer and excessive dynamic loads.

- GDC 5—The PXS is designed for a single NPP, and is not shared between units.

- GDC 17—The PXS performs its functions without relying on onsite or offsite ac power. The effects of loss of offsite power on the RCP trip and the results of the design-basis events are considered in the safety analyses which demonstrate the ability of the AP1000 to meet the acceptance criteria.

- GDC 27, 34, and 35—Safety analyses of the design-basis transients and accidents were performed with the assumption of the most reactive control rod stuck out of the core, and the results demonstrate that the PXS provides sufficient capability to remove residual heat and provide abundant core cooling so that (1) the specified acceptable fuel design limits and the design conditions of the RCS pressure boundary are not exceeded, and (2) the acceptance criteria specified in 10 CFR 50.46 for LOCAs are met.

- GDC 36 and 37—The PXS systems and components are designed to permit periodic inspection and testing of the operability of the system throughout the life of the plant.

The AP1000 design includes preoperational testing for the PXS, as discussed in DCD Tier 2, Section 14.2.9.1.3. In addition, DCD Tier 1 Information Section 2.2.3, “Passive Core Cooling System,” Table 2.2.3-4, “Inspections, Tests, Analyses, and Acceptance Criteria,” specifies (1) the design commitments of the PXS, (2) the inspections, tests, or analyses to be performed by the COL applicants, and (3) the acceptance criteria to ensure that the PXS is built by the COL applicants as designed. Therefore, the staff finds the AP1000 PXS design acceptable.

6.4 Control Room Habitability Systems

The staff reviewed the control room habitability systems in accordance with NUREG-0800, “Standard Review Plan,” Section 6.4, “Control Room Habitability System.” Conformance with the acceptance criteria of the SRP forms the basis for concluding that the control room habitability systems satisfy the following requirements:

- GDC 4, which states that structures, systems, and components important to safety shall be designed to accommodate the effects of and being compatible with the environmental conditions for normal operation, maintenance, testing, and postulated accidents

- GDC 5, regarding shared systems and components important to safety

- GDC 19, regarding providing a control room from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions

- TMI-related requirement 10 CFR 50.34(f)(2)(xxviii), regarding the evaluation of potential pathways for radioactivity and radiation that may lead to control room habitability problems

- TMI Action Plan Item III.D.3.4 (NUREG-0737), regarding protection against the effects of release of toxic substances, either on or off the site

Throughout this evaluation, reference is made to GDC 19 as applied to the AP1000 design. The staff used a dose criterion of 0.05 Sv (5 rem) total effective dose equivalent (TEDE) for evaluating the control room radiological consequences resulting from DBAs, pursuant to GDC 19 of Appendix A to CFR Part 50.

In DCD Tier 2, Section 3.1.1, Westinghouse states that the AP1000 design is a single-unit plant; if more than one unit is built on the same site, none of the safety-related systems will be shared. Thus, independence of all safety-related systems and their support systems will be maintained among the individual plants. The staff determined that the design described in the DCD does not share SSCs with other nuclear power units. Therefore, the air conditioning, HVAC systems meet the requirements of GDC 5.

During normal and postulated accident conditions, the habitability systems will provide the following:

- a controlled environment for personnel comfort and equipment operability

- radiation shielding against releases of airborne radioactive materials outside the control building

- protection against releases of airborne radioactive materials and toxic gases surrounding the control building

- protection against the effects of high-energy line ruptures in adjacent plant areas

- fire protection to ensure that the control room is manned continuously

In DCD Tier 2, Section 15.6.5.3.5, Westinghouse described the MCR dose model for calculating the radiation exposure of control room personnel for accident conditions.

The following systems provide the control room habitability functions for the plant:

- nuclear island nonradioactive ventilation system (VBS)
- main control room emergency habitability system (VES)
- radiation monitoring system (RMS)
- fire protection system (FPS)

- plant lighting system (ELS)

Section 9.5.1 of this report evaluates the use of noncombustible construction and heat and flame resistant materials throughout the plant to reduce the likelihood of fire and consequential impact on the atmosphere in the MCR envelope (MCRE). Manual hose stations outside the MCRE and portable fire extinguishers are provided to fight an MCR fire.

The RMS provides radiation monitoring and the ELS provides emergency lighting for the MCRE. The VBS provides normal and abnormal HVAC services to the MCR, technical support center (TSC), instrumentation and control rooms, dc equipment rooms, battery rooms, and the VBS equipment room, as long as an ac source of power is available. The VES is designed to provide emergency ventilation and pressurization for the MCRE when a source of ac power is not available to operate the VBS, or if radiation levels in the MCR supply air duct reach the high-high level. Section 12.3. of this report discusses radiation shielding corresponding to the design-basis LOCA. Section 15.3 of this report provides a description of design-basis LOCA source terms and an evaluation of control room operator doses. The VES is not required during normal operating conditions and is automatically initiated following a high-high particulate or iodine radioactivity signal in the MCR supply air duct, or if the VBS is inoperable (i.e., loss of ac power signals). The VES, as part of the habitability systems, is addressed in this section of this report. The VBS, FPS, ELS, and RMS are addressed in Sections 9.4.1, 9.5.1, 9.5.3, and 11.5 of this report, respectively.

The control habitability systems are capable of maintaining the MCRE environment suitable for control room operators for the duration of a postulated DBA to meet the requirements of GDC 19, as discussed in this section and in Section 15.3 of this report. Chapter 20 of this report discusses the AP1000 design's conformance with the requirements of Generic Issue B-66, "Control Room Infiltration Measurements," and TMI Action Item III.D.3.4, "Control Room Habitability."

As described in Section 9.4.1 of this report, the VBS includes redundant non-safety-related supplemental air filtration units. During abnormal operation, when high gaseous radioactivity is detected in the MCR supply air duct, and the VBS' MCR/TSC HVAC subsystem is operable, both supplemental air filtration units automatically start to pressurize the MCR/TSC areas to at least 3.2 mm (0.125 in.) water gauge using filtered makeup. Subsequently, one of the supplemental filtration units is manually shutdown. The normal outside air makeup duct and the MCR and TSC toilet exhaust duct isolation valves automatically close and the smoke/purge isolation dampers close, if open. The subsystem air handling unit continues to provide cooling in the recirculation mode to maintain the MCR/TSC areas within their design temperature. This maintains the MCRE passive heat sink below its initial ambient air design temperature in the event VES actuation is required. The supplemental filtration unit provides pressurization for the combined volume of the MCR and TSC. A portion of the recirculated air from the MCR and TSC is also filtered for clean up of airborne radioactivity.

During abnormal operation, if ac power is unavailable for more than a short period, or high-high particulate or iodine radioactivity is detected in the MCR supply air duct, which could lead to exceeding GDC 19 dose limits, the plant's safety monitoring system automatically isolates the MCRE from the normal MCR/TSC HVAC subsystem by closing the supply, return, and toilet exhaust isolation valves. The VES safety-related supply isolation valve in each train opens automatically to protect the MCRE occupants from a potential radiation release.

DCD Tier 2, Figures 6.4-1, 1.2-8, and 12.3-1 depict the MCRE. DCD Tier 2, Figures 1.2-25 through 1.2-31, illustrate the areas adjacent to the MCRE. DCD Tier 2, Table 3.2-3, indicates that the VES is located in the auxiliary building, which is a missile-protected seismic Category 1 building. The MCR pressure boundary is located on Elevation 35.81 m (117'-6") in the auxiliary building, on the nuclear island. As shown in DCD Tier 2, Figure 6.4-1, the MCRE encompasses the MCR area, tagging room, operator area, shift supervisor's office, clerk's desk, kitchen, and toilet facilities. The stairwell leading down to Elevation 30.48 m (100'-0") is not part of the MCRE.

DCD Tier 2, Sections 6.4, 9.4-1 and 15.6.5.3; Tables 6.4-1 through 6.4-3 and 15.6.5-2; and Figures 1.2-8, 1.2-25 through 1.2-31, 6.4-1, 6.4-2, and 9.4.1-1, respectively, provide the VES and interfacing VBS descriptions, design parameters, instrumentation (including indications and alarms), and figures. DCD Tier 2, Sections 7.3 and 11.5 provide details of the radiation monitors, including testing and inspection. Chapter 12 of this report evaluates the MCRE shielding design. The redundant, nonseismically qualified, and non-Class 1E-powered pressure instrumentations (PT001A/B) located outside the MCRE, as shown in DCD Tier 2, Figure 6.4-2 and Table 7.5-1, are provided to monitor the common header pressure for the VES storage tanks. The primary postaccident indications of VES operability are provided through the seismically qualified and non-Class 1E-powered differential pressure indicators and the airflow rate instrumentations.

The VES is a self-contained system with no interaction with other zones. As discussed in Section 9.4.1 of this report, normal VBS operation establishes the following conditions to ensure proper VES operation:

- The MCR/TSC HVAC subsystem maintains the MCRE and TSC between 19.4 and 22.8 °C (67 to 73 °F) and between 25 percent and 60 percent relative humidity (RH). The VBS maintains the VES passive cooling heat sink below its initial design ambient air temperature limit of 23.9 °C (75 °F).

- The Class 1E electrical room HVAC subsystem maintains the Class 1E dc equipment rooms between 19.4 and 23.9 °C (67 to 75 °F); the Class 1E electrical penetration rooms, Class 1E battery rooms, Class 1E instrumentation and control rooms, remote shutdown area, RCP trip switchgear rooms, and adjacent corridors between 19.4 and 22.8 °C (67 to 73 °F); and the HVAC equipment rooms between 10 and 29.4 °C (50 to 85 °F). The VBS maintains the Class 1E electrical room emergency passive cooling heat sink below its initial design ambient air temperature limit of 23.9 °C (75 °F).

When the VBS is not available during the 72 hours following the onset of a postulated DBA, the VES serves the function of providing passive heat sinks to limit the temperature rise in the MCR envelope, instrumentation and control rooms, and dc equipment rooms. The heat generated by the equipment, light, and occupants is absorbed by heat sinks that consist primarily of the thermal mass of the concrete that makes up the ceilings and walls of these rooms. As described in DCD Tier 2, Section 6.4.2.2, a metal form is attached to the surface of the concrete, at selected locations, to enhance the heat absorbing capacity of the ceilings. Metallic plates are attached perpendicularly to the ceiling metal form. These plates extend into the room and act as thermal fins to enhance the heat transfer from the room air to the concrete. The temperature in the instrumentation and control rooms following a loss of the VBS is limited to 48.9 °C (120 °F), and the temperature in the dc equipment rooms is limited to 48.9 °C (120 °F) because of the passive heat sinks.

The VES has two safety-related, full-capacity trains to provide emergency air pressurization of the MCRE under emergency conditions. The VES is not required to operate during normal operating conditions. The VES compressed air supply contains a set of storage tanks connected to a main and an alternate air delivery line. Components common to both lines include a manual isolation valve, a pressure regulation valve, and a flow-metering orifice. The system has sufficient redundancy to ensure operation under emergency conditions, assuming the single failure of any one component. Single-active-failure protection is provided by the use of redundant remotely operated isolation valves in the main air delivery line, which are located within the MCR pressure boundary. The Class 1E VES components are connected to independent Class 1E power supplies. Both the VES and the portions of the VBS that isolate the MCRE are designed to remain functional during an SSE or design-basis tornado. In the event of insufficient or excessive flow in the main delivery line, the main delivery line is isolated and the alternate delivery line is manually actuated by opening a manual valve that is located within the MCR pressure boundary. The alternate delivery line contains the same components as the main delivery line, with the exception of the remotely operated isolation valves, and thus is capable of supplying compressed air to the MCRE at the required flow rate.

The 32 emergency air storage tanks are constructed of forged, seamless pipe with no welds, and conform to Section VIII and Appendix 22 of the ASME Code. The design pressure of the air storage tanks is 27,600 kPa (4000 psi). DCD Tier 2, Table 3.2-3, provides data for the VES pressure-regulating valves, flow-metering orifices, remotely operated isolation valves, manual isolation valves, pressure relief isolation valves, and pressure relief dampers. The main airflow path contains a normally open, manually operated valve to isolate and preserve the contents of the air storage tanks in the event of a pressure-regulating valve malfunction. The alternate airflow path contains a normally closed, manually operated valve to manually activate the alternate delivery flowpath in the event the main delivery flow path is inoperable. The VES piping and penetrations for the MCRE are designated as safety Class C. The piping material is alloy steel (ASME Section III, Class 3, Quality Group C), except the piping from the tanks to the subheaders which is stainless steel, as shown in DCD Tier 2, Figure 6.4-2, and is corrosion resistant. Air quality testing is performed quarterly to ensure its acceptability for breathing purposes. A “pigtail” loop at the discharge side of each emergency air storage tank is provided to allow more flexibility in the connection to account for contraction and expansion in the piping. As stated in DCD Tier 2, Section 6.4.2.3, the emergency air storage tanks collectively contain a minimum storage capacity of 8,895 m^3 (314,132 ft^3) at a minimum pressure of 23,400 kPa (3,400 psig). Each pressure-regulating valve, located downstream of the common header, controls downstream pressure to approximately 790 kPa (100 psig) by means of a self-contained pressure control operator. Each flow-metering orifice provides the required flow rate to the MCRE with an upstream pressure of approximately at 790 kPa (100 psig).

Each pressure relief (butterfly) isolation valve is normally closed to prevent interference with the operation of the VBS, and provides a leaktight seal to protect the MCR pressure boundary. Each pressure relief damper, located downstream of the butterfly isolation valve, is set to open on a differential pressure of 3.2-mm (0.125-in.) water gauge with respect to its surroundings.

Two sets of doors, with a vestibule between that acts as an airlock, are provided at the access to the MCRE. The emergency exit door (to the stairs to Elevation 30.48 m (100'-0")) is normally closed, and remains closed under DBA conditions. The penetrations for the piping, ducts, conduits, and electrical cable trays through the MCRE are sealed with a seal assembly compatible with the materials of penetration commodities. The penetration sealing materials are selected to meet GDC 19 criteria for barrier and environment design and remain functional and undamaged during and following an SSE. The electrical cables are routed through internally sealed conduit. Portable, self-contained breathing equipment with air bottles to provide 6 hours of breathable air, along with a supply of protective clothing and respirators for up to 11 MCR occupants, are stored inside the MCRE.

The MCRE is designed for low-leakage construction with no-block walls. The cast-in-place reinforced concrete walls and slabs are constructed to minimize leakage through construction joints and penetrations. DCD Tier 2, Sections 3.8.4.6 and 6.4.2.4 describe the construction techniques and low-leakage features to qualify the MCRE as a low-leakage boundary. Penetration sealing materials are designed to withstand at least a 6.4-mm (0.25-in.) water gauge pressure differential in an air pressure barrier. Penetration sealing material is gypsum cement or equivalent. The non-safety-related VBS air filter housings are designed, tested, and constructed in accordance with RG 1.140, Revision 2, "Design, Inspection, and Testing Criteria for Air Filtration and Adsorption Units of Normal Atmosphere Cleanup Systems in Light-Water-Cooled Nuclear Power Plants," and ASME AG-1 and ASME N509 and N510 standards. RG 1.140, Revision 2, and ASME N509 do not allow the use of silicone sealant or any other patching material on filters, housing, mounting frames, or ducts. The non-safety-related VBS ducting is the only HVAC system ducting passing through the MCRE. It is constructed and installed in accordance with Sheet Metal and Air Conditioning Contractors' National Association (SMACNA) standards, and duct joints are sealed with qualified, nonsilicone sealant. DCD Tier 2, Section 6.4.2.4, states specifically that no silicone sealant or any other patching material is used on VBS filters, housing, mounting frame, ducts, or penetrations and VES piping, valves, dampers, or penetrations forming the MCR pressure boundary.

Westinghouse evaluated the effects of three spent fuel pool boiling scenarios on the MCRE. These scenarios consisted of station blackout (SBO) immediately following a full-core offload, an SBO concurrent with a LOCA immediately following a normal refueling, and an SBO concurrent with a LOCA 12 months following a normal refueling. The evaluation results showed that the temperature for the personnel access route and the safety-related valve areas remained below 43 °C (110 °F) (initial temperature of 40 °C (104 °F)) for at least 72 hours after the event and, therefore, the accessibility and equipment qualification were not challenged. DCD Tier 2, Section 6.4.2.4, states that no adverse environmental effects will occur to the MCR sealant materials resulting from postulated spent pool boiling events.

DCD Tier 2, Section 6.4.3.2, states criteria for meeting MCRE air contaminants, including carbon dioxide requirements. The applicant also evaluated both equipment and human performance against the effects of the highest humidity in the MCRE. Westinghouse performed an evaluation using the GOTHIC code and MCRE moisture balance, with respect to time, for a maximum of 11 MCR occupants, during the first 72 hours of an accident. With initial conditions of 24 °C (75 °F) and 60 percent RH, the thermal analysis resulted in the following:

- 31 °C (87 °F) and 41 percent RH at 3 hours, when the non-Class 1E battery heat loads are exhausted
- 29 °C (84 °F) and 45 percent RH at 24 hours, when the battery heat loads are terminated
- 30 °C (86 °F) and 39 percent RH at 72 hours

The staff finds that the above results are within the guidelines of MIL-HDBK-759C, 31 July 1995, “Human Engineering Design Guidelines,” and MIL-STD-1472E, 31 October 1996, “Human Engineering.”

DCD Tier 2, Section 6.4.4, states that the VES nominally provides 104.5 standard cubic meters per hour (scmh) (65 standard cubic feet per minute (scfm)) of ventilation air to the MCRE from the air storage tanks through the main or alternate air delivery line. Westinghouse also states in this section that the VES flow of 96.5 scmh (60 scfm) is sufficient to pressurize the MCRE to at least (positive) 3.2-mm (0.125-in.) water gauge differential pressure (with respect to the surroundings), and to maintain carbon dioxide concentration below 0.5 percent by volume for a maximum occupancy of 11 persons inside the MCRE. This will maintain air quality within the guidelines of Table 1 and Appendix C, Table C-1, to American Society of Heating, Refrigerating, and Air-Conditioning Engineers (ASHRAE) Standard 62-1989, “Ventilation for Acceptable Indoor Air Quality.” Westinghouse's latest leak-rate analysis assumes a MCRE occupancy limited to 11 persons throughout the 72-hour period following an accident and is predicated on the validation process task analysis described in DCD Tier 2, Chapter 18.

The safety-related compressed air storage tanks are sized to provide the airflow to the MCRE for 72 hours. During a nonradiological emergency, the emergency air storage tanks can be refilled via a connection to the breathable quality air compressor in the compressed and instrument air system (CAS). These tanks can also be refilled from portable supplies by an installed connection in the CAS.

DCD Tier 2, Section 6.4.4, states that the analysis of onsite chemicals is described in DCD Tier 2, Table 6.4-1, and their locations are shown in DCD Tier 2, Figure 1.2-2. Analysis of these sources is in accordance with RG 1.78, Revision 1, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release,” and shows that these sources do not represent a toxic hazard to MCRE personnel.

The NRC staff requested additional information as part of the RAI 410.007 to (a) verify that the chemicals listed in DCD Tier 2, Table 6.4-1, “Onsite Chemicals,” were evaluated using the methodology in NUREG-0570, “Toxic Vapor Concentrations in the Control Room Following a

Postulated Accidental Release," and that these chemicals do not represent a toxic hazard to control room operators; (b) verify that COL applicants are responsible for (i) the amount and location of possible sources of toxic chemicals (as shown in DCD Tier 2, Table 6.4-1, and their locations, as shown in DCD Tier 2, Figure 1.2-2) in or near the plant, (ii) seismic Category I Class 1E toxic gas monitoring, as required, (iii) assessing control room protection for toxic chemicals, and (iv) evaluating offsite toxic releases (including the potential for toxic releases beyond 72 hours) in accordance with RG 1.78, Revision 1, and meet the requirements of 10 CFR 50.34(f)(2)(xxviii) (TMI Action Plan Item IIID.3.4) and GDC 19; add RG 1.78, Revision 1, to DCD Tier 2, Section 6.4.8, "References" because RG 1.78, Revision 1, replaces both RG 1.78, Revision 0, and RG 1.95, Revision 1, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release"; (d) delete reference to RG 1.95 from DCD Tier 2, Section 6.4.7; (e) revise Appendix 1A to assess the conformance with RG 1.78, Revision 1, and revise DCD Tier 2, Sections 2.2, 6.4, 9.4.1, and 9.5.1, and Table 1.9-1 (Sheet 7 of 15) to correctly state the reference as "RG 1.78 December 2001, Revision 1"; and (f) revise the reference list in TS bases B.3.7.6 to add a reference to ASHRAE Standard 62-1989.

In a letter dated March 26, 2003, Westinghouse revised the response to RAI 410.007 by providing additional information as requested by the NRC staff and committing to revise DCD Tier 2, Sections 6.4.4, 6.4.7, and 6.4.8; Appendix 1A to DCD Tier 2; and DCD Tier 2, Chapter 16, B3.7.6. Westinghouse incorporated these changes in the DCD. However, the staff noted that the DCD still needed to include the response to RAI 410.007(a), Revision 2, dated March 26, 2003. Specifically, the DCD needed to include a statement that Westinghouse had verified that the chemicals listed in DCD Tier 2, Table 6.4-1, "Onsite Chemicals," were evaluated using the methodology in NUREG-0570 and concluded that these chemicals do not represent a toxic hazard to control room operators. In a letter dated May 21, 2003, Westinghouse revised the response to RAI 410.007 and committed to placing this information in the DCD. This was Confirmatory Item 6.4-1 in the DSER. In a subsequent DCD revision, the staff verified that Westinghouse included this statement. Therefore, Confirmatory Item 6.4-1 is closed.

The staff performed an independent evaluation. On the basis of the data Westinghouse furnished regarding quantity, sizes, and locations, the staff concludes that these onsite chemicals meet the guidelines of RG 1.78, Revision 1.

In DCD Tier 2, Section 6.4.7, Westinghouse states that COL applicants referencing the AP1000 design are responsible for the amount and location of possible sources of toxic chemicals in or near the plant, as well as for seismic Category 1, Class 1E toxic gas monitoring, as required (detectors where necessary to permit automatic isolation of the control room). Additionally, it further states that RG 1.78, Revision 1, addresses control room protection for toxic chemicals, and that Westinghouse evaluated offsite releases (including the potential for toxic releases beyond 72 hours in accordance with the guidelines of RG 1.78) in order to meet the requirements of the TMI Action Plan Item III.D.3.4 and GDC 19.

As discussed previously, the non-safety-related VBS subsystem (MCR/TSC HVAC subsystem) isolates the MCRE and/or TSC area from the normal outdoor air intake. It provides filtered outdoor air to pressurize the MCRE and TSC areas to a positive pressure of at least 3.2-mm (0.125-in.) water gauge, with respect to the surrounding areas, when high gaseous radioactivity is detected in the MCRE supply air duct. The non-safety-related supplemental air filtration units have a fission product removal efficiency of 90 percent for charcoal adsorbers and 99 percent for high-efficiency particulate air (HEPA) filters.

No credit was taken for fission product removal by HEPA filters and charcoal adsorbers in the supplemental air filtration units in evaluating the control room radiological habitability.

The VBS system is not designed as a postaccident ESF atmospheric cleanup system and has no safety-grade source of power; therefore, it was not credited in evaluating conformance with GDC 19. Section 9.4 of this report provides the staff's evaluation of the VBS.

The location of the single control room outside air intake serving the VBS conforms with the guidance of Section 6.4 of the SRP and RG 1.95 because it is located more than 15.2 m (50 ft) vertically below and more than 30.5 m (100 ft) laterally away from the plant discharge. The air intake is located on the roof of the auxiliary building at Elevation 46.63 m (153'-0"), and is protected by an intake enclosure. The VBS redundant radiation monitors are located inside the MCRE. DCD Tier 2, Figure 9.4.1-1, depicts the radiation monitors and outside air isolation dampers. The outside air is continuously monitored by redundant smoke monitors at the outside air intake. As stated in DCD Tier 2, Section 9.4.1.2.1.1, the VBS supply, return, and toilet exhaust ducts are the only HVAC penetrations in the MCRE; as stated in DCD Tier 2, Section 6.4.4, no radioactive materials are stored or transported near the MCRE.

The flue gas exhaust stacks of the onsite standby power diesel generators are located in excess of 46 m (150 ft) away, and the onsite standby power system fuel oil storage tanks are located in excess of 91 m (300 ft) away from the fresh air intakes of the MCR to preclude the drawing of combustion fumes or smoke from an oil fire into the MCR.

GDC 19 requires that the control room be designed to provide adequate radiation protection permitting personnel to access and occupy the control room under accident conditions. As applied to the AP1000 design, GDC 19 requires that adequate radiation protection be provided to ensure that radiation exposures will not exceed 0.05 Sv (5 rem) TEDE, as defined in 10 CFR 50.2, for the duration of the accident. Westinghouse proposed that this requirement be met by incorporating sufficient shield walls and by installing the redundant non-safety-related supplemental air filtration units (VBS) and a safety-related emergency bottled air pressurization system (VES). Section 9.4 of this report provides a discussion of the staff's review of the applicant's analysis of the capability of the non-safety-related VBS to mitigate the consequences of a design-basis accident in the MCR and TSC.

Westinghouse submitted the results of radiological consequence analyses for personnel in the MCR during design-basis accidents in DCD Tier 2, Section 6.4.4. Details of the analysis assumptions for modeling the doses to MCR personnel were submitted in DCD Tier 2, Section 15.6.5.3. Section 15.3 of this report discusses the staff's review of the applicant's analysis.

To verify the Westinghouse assessments, the staff performed independent radiological consequence calculations for DBAs with the VES under high-high radiation level as described in the DCD Tier 2, Section 6.4. The staff used the following information in its analyses:

- reactor accident source terms based on NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” and RG 1.183, “Alternative Radiological Source Terms for Evaluating Design-Basis Accidents at Nuclear Power Reactors”

- control room χ/Q values and control room unfiltered in-leakage rates provided by Westinghouse

- control room occupancy factors referenced in Section 6.4 of the SRP

Because of issues with aerosol removal and atmospheric dispersion as discussed in DSER Open Items 15.3-1 and 15.3-2, the staff was unable to complete its review of the applicant’s analyses of the radiological consequences in the MCR during design-basis accidents. This was Open Item 6.4-1 in the DSER. With the resolution of DSER Open Items 15.3-1 and 15.3-2, the staff has completed its review, and DSER Open Item 6.4-1 is closed.

The staff finds that the VES, under high-high radiation conditions, as described in DCD Tier 2, Section 6.4, is capable of mitigating the dose in the MCR following DBAs to meet the dose criteria specified in GDC 19, as applied to the AP1000 design. Section 15.3 of this report discusses the staff’s review of the applicant’s analysis of control room habitability and the staff’s independent confirmatory radiological consequence analyses for the control room operators.

DCD Tier 2, Section 6.4.7, states that COL applicants referencing the AP1000 certified design are responsible for verifying that the procedures and training for control room habitability are consistent with the intent of GSI 83 (see DCD Tier 2, Section 1.9). This is COL Action Item 6.4-1.

The VES is tested and inspected at appropriate intervals, in accordance with the surveillance and frequency requirements specified in the TS. The leaktightness testing of the MCRE is conducted in accordance with the frequency specified in the TS. Connections are provided for sampling the air supplied from the CAS and for periodic sampling of the air stored in the emergency air storage tanks. In accordance with the TS, air samples from the emergency air storage tanks are taken quarterly (every 92 days) and analyzed to ensure conformance with the guidelines of Table 1 and Appendix C, Table C-1, of ASHRAE Standard 62-1989.

DCD Tier 2, Table 15.6.5-2, provides the MCRE volume and maximum unfiltered air in-leakage (infiltration) rates as follows. The MCRE volume is 1,011 m^3 (35,700 ft^3). The maximum unfiltered air in-leakage (infiltration) into the MCRE under accident conditions is 4.02–8.04 scmh (2.5–5.0 scfm) when the VES is operating. The maximum unfiltered air in-leakage (infiltration) into the MCRE during a high gaseous radioactivity signal while the VBS is operating is 145 scmh (90 scfm). The AP1000 design includes an air-lock type, double door vestibule style entrance for the MCRE to minimize contaminated air from entering the MCRE as a result of egress and ingress, and to maintain the MCRE at 3.2-mm (0.125-in.) water gauge positive pressure, with respect to surrounding areas.

DCD Tier 2, Section 6.4.5.4, states that “Testing for main control room in-leakage during VES [main control room emergency habitability system] operation will be conducted in accordance with ASTM E741 [2000, “Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution”].

In RAI 410.007, the NRC staff stated that it anticipates that the testing frequency for air in-leakage will be 5 to 6 years, based on joint efforts currently pursued by the industry and NRC staff to address control room habitability issues. Therefore, the AP1000 design should include a commitment to resolving the in-leakage testing frequency issue in accordance with the anticipated outcome of the joint effort between the NRC staff and industry.

In a letter dated November 15, 2002, Westinghouse responded that the NRC staff and the industry are working on in-leakage testing; however, it is not reasonable to commit to a standard that does not currently exist. Westinghouse, therefore, is not providing a commitment to have the VES meet the anticipated requirements currently being pursued. Westinghouse further stated that the VES design addresses in-leakage and meets the codes and standards that were in effect 6 months prior to the date of the AP1000 design certification application (March 28, 2002). The NRC staff disagreed with Westinghouse’s position on the testing frequency for unfiltered-in-leakage, as provided in its response to RAI 410.007, and stated that Westinghouse needs to revise its RAI 410.007 response and DCD Tier 2, Section 6.4.5.4, to provide an in-leakage testing frequency commitment commensurate with the anticipated outcome of the joint effort between the NRC staff and industry. In a letter dated February 14, 2003, Westinghouse provided additional information to revise its original response to RAI 410.007 asserting that DCD Tier 2, Section 6.4.5.4, will be revised to state that, “Testing for main control room inleakage during VES operation will be conducted in accordance with ASTM E741"; DCD Tier 2, Section 6.4.7, will be revised to state that, “The Combined License applicant will provide the testing frequency for the main control room inleakage test discussed in DCD Tier 2, Section 6.4.5.4.” In addition, Westinghouse revised DCD Tier 2, Table 1.8-2, “Summary of AP1000 Standard Plant Combined License Information Items,” Item 6.4-3, to refer to DCD Tier 2, Section 6.4.7 for the MCR in-leakage test frequency. Westinghouse incorporated these changes into the DCD. This is COL Action Item 6.4-2; the staff finds this approach to be acceptable because the COL applicant will actually measure the control room unfiltered inleakage at a frequency that will be reviewed by the staff at the COL stage.

DCD Tier 2, Section 6.4.2.2, states that, in the unlikely event that power to the VBS is not available for more than 72 hours and the outside air is acceptable radiologically and chemically, MCR habitability is maintained by operating one of the two MCR ancillary fans to supply outside air to the MCR. Doors and ducts may be opened to provide a supply pathway and an exhaust pathway for the ancillary fans. Likewise, outside air is supplied to Divisions B and C instrumentation and control rooms to maintain the ambient temperature below the qualification temperature of the equipment. It is expected that outside air will be acceptable within 72 hours following a radiological and toxic gas release. The outside air pathway to the ancillary fans is provided through the VBS air intake opening located on the roof, the mechanical room at floor Elevation 41.22 m (135'-3"), and the VBS supply duct. Warm air from the MCRE is vented to the annex building through stairway S05, and into the remote shutdown room and the clean access corridor at Elevation 30.48 m (100'-0"). As stated in DCD Tier 2, Section 9.4.1.1.2, the post-72 hour design basis of the VBS is (1) to maintain the MCR below a temperature approximately 2.5 °C (4.5 °F) above the average outdoor air temperature, and (2) to maintain Divisions B and C instrumentation and control rooms below the qualification temperature of the instrumentation and control equipment. Section 8.3 of this report discusses the staff's evaluation of the post-72 hour power supply.

Chapter 14 of this report discusses preoperational testing. It includes verification that a minimum VES airflow rate of 104.5±8.04 scmh (65±5 scfm) will pressurize the MCRE to 3.2-mm (0.125-in.) water gauge with respect to the surrounding spaces. The maximum unfiltered air in-leakage (infiltration) rate of 4.02–8.04 scmh (2.5–5.0 scfm) during accident conditions when the VES is in operation will be verified in accordance with ASTM E741, "Standard Test Method for Determining Air Change in a Single Zone by Means of a Tracer Gas Dilution." The 72-hour capacity of air storage tanks will be verified to be in excess of 8,418 standard cubic meters (314,132 standard cubic feet), at a minimum pressure of 23,442 kPa (3,400 psig). Heat loads will be verified to be below the values in DCD Tier 2, Table 6.4-3. VBS MCRE isolation valves will be tested to verify the leaktightness of the valves. Section 11.5 of this report discusses testing and inspection of the VBS safety-related radiation monitors. The air quality within the MCR/TSC environment will be confirmed to be within the guidelines of Table C-1 of ASHRAE Standard 62-1989 by analyzing air samples taken during pressurization testing. The staff finds the preoperational testing to be acceptable because it will verify the ability of the MCRE to limit unfiltered in-leakage and maintain acceptable air quality and a suitable environment for the operators.

The VES indications and alarms listed in DCD Tier 2, Table 6.4-2, are located in the MCR. Sections 7.3 and 11.5 of this report discuss actuation and radiation monitoring instrumentation for the VBS and VES.

Westinghouse evaluated the MCRE structure for protection against the environmental requirements, including soil and water pressure, on substructure, tornado pressure drop, thermal stresses, and pipe and pipe rupture loads in DCD Tier 2, Sections 3.3, 3.6, and 3.8. Westinghouse also stated that the flood protection measures for seismic Category 1 SSCs are designed in accordance with RG 1.102, "Flood Protection for Nuclear Power Plants," and RG 1.59, "Design Basis Floods for Nuclear Power Plants." Additionally, Westinghouse states the following in DCD Tier 2, Sections 3.5 and 3.6:

- Internally generated missiles (outside the containment) from rotating and pressurized components are either not considered credible or evaluated as described in DCD Tier 2, Section 3.5.1.1.

- Protection from high-energy lines near the control room is evaluated in DCD Tier 2, Section 3.6.1.2.

Therefore, Westinghouse concludes that the habitability systems will be protected against dynamic effects that may result from possible failures of such lines.

In Sections 3.4.1, 3.5.1.1, 3.5.2, and 3.6.1 of this report, the staff documents its evaluation of the protection against floods, internally and externally generated missiles, and high- and moderate-energy pipe breaks. The staff concludes that the control room habitability systems satisfy GDC 4, as it relates to protection of the system against floods, internally generated missiles, and piping failures.

As described above, the staff evaluated the VES for conformance with GDC 4, 5, and 19, as referenced in Section 6.4 of the SRP, and consequently with the subject SRP acceptance criteria. The staff finds the VES acceptable.

Control Room Habitability and Toxic Chemicals

Westinghouse specifies in DCD Tier 2, Section 6.4.7, that the evaluation of possible harmful effects to control room personnel from toxic chemicals located at or near the site will be addressed by the COL applicant. The staff finds this acceptable. This is COL Action Item 6.4-3.

6.5 Fission Product Removal and Control Systems

6.5.1 ESF Plant Atmosphere Filtration Systems

This section is not applicable to the AP1000 design.

6.5.2 Containment Spray System

The AP1000 design does not have a safety-related containment spray system. Its design involves removal of airborne activity by a natural process that does not depend on sprays (i.e., sedimentation, diffusiophoresis, and thermophoresis). Much of the nongaseous airborne activity would eventually be deposited in the containment sump solution. Long-term retention of iodine in the containment sump following DBAs requires adjustment of the sump's pH. For the AP1000 design, this adjustment is accomplished through the PXS discussed in DCD Tier 2, Section 6.3, "Passive Core Cooling System." The FPS provides a non-safety-related containment spray function for accident management following a severe accident. This design is not credited in any analysis. DCD Tier 2, Section 15.6.5, "Loss-of-Coolant Accidents Resulting from a Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary," further discusses natural mechanisms for removal of airborne activity.

The AP1000 design does include a non-safety-related containment spray as part of the FPS, and which is used to enhance the natural removal mechanisms in the unlikely event of a severe accident. The containment isolation portion of this system is safety-related. Section 19.2.3.3.9 of this report evaluates the non-safety-related containment spray system.

6.5.3 Fission Product Control Systems

The AP1000 has no active system to control fission products in the containment following a postulated accident. The only fission product control system is the primary containment. DCD Tier 2, Appendix 15B discusses satisfactory removal of airborne activity (elemental iodine and particulates) from the containment atmosphere by natural removal processes (e.g., deposition and sedimentation) without the use of containment spray. The AP1000 design does not require active fission product control systems to meet the regulatory requirements (dose limits in 10 CFR 50.34). These natural fission product control mechanisms and the limited containment leakage result in offsite doses that are less than those specified in 10 CFR 50.34.

6.6 Inservice Inspection of Class 2 and 3 Components

The staff reviewed DCD Tier 2, Section 6.6, “Inservice Inspection of Class 2 and 3 Components,” in accordance with Section 6.6, “Inservice Inspection of Class 2 and 3 Components,” of the SRP. The SRP, Section 6.6, states that the requirements for periodic inspection and testing of Class 2 and 3 systems in GDC 36, 37, 39, 40, 42, 43, 45, and 46, are specified in 10 CFR 50.55a and detailed in Section XI of the ASME Code.

The ISI program for ASME Class 2 and Class 3 components relies upon these design provisions to allow performance of ISI. Compliance with these GDC ensures that the design of the safety systems will allow accessibility of important components so that periodic inspections can be performed to detect degradation, leakage, signs of mechanical or structural distress caused by aging, and fatigue or corrosion, prior to the ability of the systems to perform their intended safety functions being jeopardized.

GDC 36 requires that the ECCS be designed to permit periodic inspection of important components to assure the integrity and capability of the system.

GDC 37 requires that the ECCS be designed to permit periodic pressure testing to assure the structural and leaktight integrity of its components.

GDC 39 requires that the containment heat removal system be designed to permit periodic inspection of important components to assure the integrity and capability of the system.

GDC 40 requires that the containment heat removal system be designed to permit periodic pressure testing to assure the structural and leaktight integrity of its components.

GDC 42 requires that the containment atmosphere cleanup systems be designed to permit periodic inspection of important components to assure the integrity and capability of the systems. As discussed below, this criterion is not applicable to the AP1000 design.

GDC 43 requires that the containment atmosphere cleanup systems be designed to permit periodic pressure testing to assure the structural and leaktight integrity of their components. As discussed below, this criterion is not applicable to the AP1000 design.

GDC 45 requires that the cooling water system be designed to permit periodic inspection of important components to assure the integrity and capability of the system. As discussed below, this criterion is not applicable to the AP1000 design.

GDC 46 requires that the cooling system be designed to permit periodic pressure testing to assure the structural and leaktight integrity of its components. As discussed below, this criterion is not applicable to the AP1000 design.

Compliance with the preservice and inservice examination requirements of 10 CFR 50.55a, as detailed in Section XI of the ASME Code, constitutes an acceptable basis for satisfying, in part, the requirements of GDC 36, 37, 39, 40, 42, 43, 45, and 46. Subsection II of the SRP states that GDC 36, 37, 39, 40, 42, 43, 45, and 46 require that the respective safety systems addressed by these criteria be designed such that they permit periodic inspection, pressure testing, and functional testing of system components and piping.

The following six specific requirements apply to the review of DCD Tier 2, Section 6.6:

(1) Components Subject to Inspection

The applicant's definition of ASME Code Class 2 and 3 components and systems subject to an inservice inspection (ISI) program is acceptable if it is in agreement with the definitions of ASME Code, Section III, Article NCA-2000, "Classification of Components and Supports," which is invoked by 10 CFR 50.55a.

(2) Accessibility

As required by 10 CFR 50.55a(g)(3)(ii), ASME Code Class 2 and Class 3 components and supports must be designed and provided with access to enable the performance of inservice examination of such components and to meet the preservice examination requirements set forth in ASME Section XI. ASME Section XI, Subarticle IWA-1400(b), states that it is the owner's responsibility for the design and arrangement of system components to include allowances for adequate access and clearances to conduct examinations and tests. ASME Section XI, Subarticle IWA-1500, establishes the requirements for accessibility in order to facilitate examination of components.

Provisions for accessibility must include (a) access for the inspector, examination personnel, and equipment necessary to conduct the examinations, (b) sufficient space for removal and storage of structural members, shielding, and insulation, (c) installation and support of handling machinery where required to facilitate removal, disassembly, and storage of equipment, components, and other materials, (d) performance of examinations alternative to those specified in the event structural defects or indications are revealed that may require such alternative examination, and (e) performance of necessary operations associated with repairs or installation of replacements.

(3) Examination Categories and Methods

The applicant's examination categories and methods of examination are acceptable if they are in agreement with the requirements of IWA-2000, IWC-2000, and IWD-2000 ("Examination and Inspection") of Section XI of the ASME Code.

(4) Evaluation of Examination Results

The methods for evaluation of the results are acceptable if they are in agreement with the requirements of IWC-3000 and IWD-3000 ("Acceptance Standards") of Section XI of the ASME Code.

(5) System Pressure Tests

The system pressure testing is acceptable if it meets the requirements of IWA-5000, “System Pressure Tests,” of Section XI of the ASME Code.

(6) Augmented ISI to Protect against Postulated Piping Failure

High-energy fluid piping between containment isolation valves receives an augmented 100-percent volumetric examination of circumferential and longitudinal pipe welds in accordance with the guidance of SRP 3.6.2, “Determination of Rupture Locations and Dynamic Effects Associated with Postulated rupture of Piping.”

Summary of Technical Information

DCD Tier 2, Section 6.6, “ISI of Class 2 and 3 Components,” indicates preservice, ISI, and testing of ASME Code Class 2 and 3 components are performed in accordance with Section XI of the ASME Code, including addenda required by 10 CFR 50.55a(g). This includes all ASME Code Section XI mandatory appendices.

The inspection program should delineate the specific edition and addenda of the Code used to determine the requirements for the inspection and testing plan for the initial and subsequent inspection intervals. DCD Tier 2, Section 5.2.1.1, indicates the baseline used for the evaluation done to support the safety analysis report and the Design Certification is the 1998 Edition through the 2000 Addenda. The Code includes requirements for system pressure tests for active components. Section XI, IWA-5000, defines the requirements for system pressure tests and visual examinations. These tests verify the pressure boundary integrity in conjunction with ISI.

Westinghouse stated that ASME Code Class 2 and 3 components are designed so that access is provided in the installed condition for visual, surface, and volumetric examinations specified by the ASME Code. Westinghouse stated that design provisions, in accordance with ASME Section XI, IWA-1500, are formally implemented in the Code Class 2 and 3 component design process. Removable insulation is provided on piping systems requiring volumetric and surface inspection. Removable hangers and pipe whip restraints are provided, where practical and necessary, to facilitate ISI. Working platforms are provided in areas requiring inspection and servicing of pumps and valves. Temporary or permanent platforms, scaffolding, and ladders are provided to facilitate access to piping welds. The components and welds requiring ISI are designed to allow for the application of the required ISI methods. Westinghouse stated that sufficient clearances for personnel and equipment, maximized examination surface distances, two-sided access, favorable materials, weld joint simplicity, elimination of geometrical interferences, and weld surface preparation all contribute to satisfying the inspectability and accessibility requirements of 10 CFR 50.55a(g)(3)(ii) and ASME Section XI, Subarticle IWA-1500. Westinghouse stated that space is provided to handle and store insulation, structural members, shielding, and other material related to the inspection. Suitable hoists and other handling equipment, lighting, and sources of power for inspection equipment are installed at appropriate locations.

Westinghouse also stated that COL applicants referencing the AP1000 certified design will prepare a preservice inspection program and an ISI program for ASME Code, Section III, Class 2 and 3 systems, components, and supports. The preservice/ISI programs will address the equipment and techniques used. This is COL Action Item 6.6-1. Finally, COL applicants referencing the AP1000 certified design will address the controls to preserve accessibility and inspectability for ASME Code, Section III, Class 2 and 3 components and piping during construction or other postdesign certification activities. This is COL Action Item 6.6-2. The preservice/ISI programs will comply with applicable provisions of 10 CFR 50.55a(b)(2).

Staff Evaluation

The staff's evaluation of ISI of ASME Code Class 2 and 3 components is divided into the following seven sections: components subject to inspection, accessibility, examination categories and methods, evaluation of examination results, system pressure tests, augmented ISI to protect against postulated piping failure, and GDC.

(1) Components Subject to Inspection

The AP1000 design classifies components as ASME Code Class 2 and 3 in accordance with the criteria provided in DCD Tier 2, Section 3.2.2, and reviewed in the corresponding section of this report. The design follows ASME Code, Section III, as required by 10 CFR 50.55a. Thus, Class 2 and 3 components subject to inspection are in agreement with definitions acceptable to the staff in ASME Code, Section III, Article NCA-2000.

(2) Accessibility

The AP1000 design follows the ASME Code provisions for accessibility which include (a) access for the inspector, examination personnel, and equipment necessary to conduct the examinations, (b) sufficient space for removal and storage of structural members, shielding, and insulation, (c) installation and support of handling machinery where required to facilitate removal, disassembly, and storage of equipment, components, and other materials, (d) performance of examination alternatives to those specified in the event structural defects or indications are revealed that may require such alternative examination, and (e) performance of necessary operations associated with repairs or installation of replacements. As required by 10 CFR 50.55a, the design of the pressure-retaining components meets the requirements of ASME Code, Section XI, Section IWA-1500, "Accessibility," thus meeting requirements acceptable to the staff with respect to accessibility.

The staff reviewed DCD Tier 2, Section 6.6, to assure that compliance with the regulations for the design of the ASME Code Class 2 and Class 3 components would be met. The regulations and the ASME Code require that inspectability and accessibility be designed into the system in order that meaningful preservice and ISIs can be performed prior to and during the life of the plant. If the provisions allowing for inspection and access for performance of preservice and ISI are not designed into the plant, the COL applicant will not be able to perform the required testing. This testing is necessary to assure that the components can perform their intended functions and do not degrade due to service-related failures.

The Westinghouse AP1000 design incorporates lessons learned to ensure that the ASME Code Class 2 and 3 components are designed to allow for the application of the required ISI methods (i.e., maximized examination surface distances, elimination of geometric interferences, weld joint simplicity, favorable materials, proper weld surface preparation, removable insulation, two-sided access, and removable whip restraints and hangers) to facilitate access for the performance of inspection. According to staff experience, these aspects of the design have been the major source of licensees' requests for relief from the ASME Code. Westinghouse also stated that access for testing, by designing sufficient platforms, and lighting, as well as installing temporary platforms and ladders to allow inspection of piping and welds, is inherent in the AP1000 design.

By effectively eliminating these interferences by designing for inspectability and accessibility, the AP1000 design meets the requirements of 10 CFR 50.55a(g)(3)(ii) and ASME IWA-1500, which enables the COL applicant to perform preservice and ISIs, and is, therefore, acceptable. The applicant has stated that relief from Section XI requirements will not be required for ASME Code, Section III, Class 2 and 3 pressure-retaining components in the AP1000 plant for the baseline design certification Code. Future changes in the Section XI requirements could, however, necessitate relief requests. The staff concludes that this approach is consistent with the requirements of 10 CFR 50.55a, and is therefore acceptable.

(3) Examination Categories and Methods

The ISI program will follow ASME Code, Section XI, as required by 10 CFR 50.55a. Thus, the examination categories and methods will be in agreement with requirements acceptable to the staff in IWA-2000, IWC-2000, and IWD-2000 of Section XI of the Code. The staff will review the ISI program at the COL stage to ensure that it meets the applicable requirements of the ASME Code, Section XI.

(4) Evaluation of Examination Results

The ISI program will follow ASME Code, Section XI, as required by 10 CFR 50.55a. Thus, the evaluation of examination results will be in agreement with requirements acceptable to the staff in IWC-3000 and IWD-3000 of Section XI of the Code. The staff will review the ISI program at the COL stage to ensure that the examination results will be evaluated in accordance with the applicable requirements of the ASME Code, Section XI.

(5) System Pressure Tests

The ISI program will follow ASME Code, Section XI, as required by 10 CFR 50.55a. Thus, the system pressure testing will meet requirements acceptable to the staff in IWA-5000 of Section XI of the Code. The staff will review the ISI program at the COL stage to ensure that the system pressure test requirements of the ASME Code, Section XI will be met.

(6) Augmented ISI to Protect against Postulated Piping Failure

DCD Tier 2, Section 6.6, indicates that the COL applicant will develop an augmented inspection program for high-energy fluid system piping between containment isolation valves. Such a program is also developed for those cases in which no isolation valve is used inside containment between the first rigid pipe connection to the containment penetration, or the first pipe whip restraint inside containment, and the outside isolation valve. This program will provide for 100-percent volumetric examination of circumferential and longitudinal pipe welds during each inspection interval conducted according to the ASME Code, Section XI. This program will cover the break exclusion portion of the high-energy fluid systems described in DCD Tier 2, Sections 3.6.1 and 3.6.2. Because the proposed program satisfies the criteria of SRP Section 6.6, the staff finds this augmented ISI program to be acceptable.

(7) GDC

The applicability of the GDC was reviewed for the AP1000 design. Because of the passive design concepts of the AP1000 design, portions of systems that had been considered safety-related in existing LWR designs and evolutionary plants are not necessarily safety-related in the AP1000 design. Consequently, these systems, or portions thereof, are not classified as ASME Code Class 2 or 3 systems; rather, they are classified as non-ASME Code systems. As non-ASME Code systems, they are not subject to ISI and periodic pressure testing required by the ASME Code. The staff, therefore, reviewed the applicability of the above GDC as they relate to the periodic inspection and testing of those portions of the ECCS, containment heat removal system, containment atmosphere cleanup system, and cooling water system that exist in the AP1000 design.

Emergency core cooling is performed by the AP1000 PXS, as described in DCD Tier 2, Section 6.3. Section 6.3 of this report describes the staff’s evaluation of the use of the PXS in lieu of an ECCS. This system is safety-related and contains ASME Code Class 1, 2, and 3 components. As such, this system is subject to the periodic inspection and pressure testing required by the ASME Code. This system is designed to permit periodic inspection and testing of components. Thus, the staff finds that the PXS meets the requirements of GDC 36 and 37.

Containment heat removal is performed by the PCS, as described in DCD Tier 2, Section 6.2.2. The PCS utilizes the steel containment shell to transfer heat from the interior through natural convection. Heat is removed from the shell by a direct-flow natural convection design and a passive external cooling system. Section 6.2.2 of this report discusses the staff’s evaluation of the PCS. This system is safety-related and contains ASME Code Class 3 components. As such, this system is subject to the periodic inspection and pressure testing required by the ASME Code. The system piping and components are designed to permit access for periodic inspection and testing of equipment. Thus, the staff finds the PCS meets the requirements of GDC 39 and 40.

The AP1000 design does not use a containment atmosphere cleanup system, as found in existing LWRs. The AP1000 does not rely on active systems for the removal of activity from the containment atmosphere postaccident cleanup functions. The containment atmosphere is depleted of elemental iodine and of particulates as a result of natural processes within containment. However, a portion of the FPS that serves a non-safety-related containment spray function for severe accident management includes equipment and valves, such as the fire pumps and fire main header. Section 6.5.2 and Chapter 19 of this report provide the staff’s evaluation of the containment spray system. Because the containment spray system has no safety function, the system components are not classified as ASME Code class, except for those portions that function as containment isolation. Those portions are classified as ASME Code Class 2. As such, no periodic inspection and pressure testing requirements apply, except for those portions of the containment spray system classified as ASME Code Class 2. The staff finds that ASME Code, Section XI, inspection and testing of a containment atmosphere cleanup system, as provided by the containment spray system, are not required because the safety-related functions of the containment atmosphere cleanup do not rely on active systems. Therefore, GDC 42 and 43 are not applicable to the AP1000 design.

The AP1000 design utilizes a component cooling water system to support the normal operation of safety-related components. However, none of the safety-related components require cooling water to perform their safety-related function. Safety-related cooldown and decay heat removal functions are provided by the PXS and the PCS. Section 9.2.2 of this report discusses the staff's evaluation of the component cooling water system. Because this system is not safety-related, the system components are not classified as ASME Code Class 1, 2, or 3, except for those portions that function as containment isolation. These system components are classified as ASME Code Class 2. As such, no periodic inspection and pressure testing requirements apply, except for those portions classified as ASME Code Class 2. The staff finds that ASME Code, Section XI, inspection and testing of the component cooling water system are not required because the safety-related functions of the component cooling water system are subsumed by the passive systems discussed above. Therefore, GDC 45 and 46 are not applicable to the AP1000 design.

Conclusions

The staff concludes that the AP1000 ISI program for Code Class 2 and 3 components is acceptable and meets the inspection and pressure-testing requirements of GDC 36, 37, 39, and 40, as well as the requirements of 10 CFR 50.55a with regard to preservice and inservice inspectability of these components.