6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

6.1.1 Structural Materials

The staff reviewed 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 (LOCA). 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 Appendix A to Title 10 of the Code of Federal Regulations (CFR) Part 50, Appendix B of 10 CFR Part 50, and 10 CFR Part 50.55a apply to systems provided to serve as ESF systems.

GDC 1 of Appendix A to 10 CFR Part 50 and 10 CFR 50.55a(a)(1) require that systems, structures, and components (SSCs) important to safety be designed, fabricated, erected, and tested to quality standards that shall be identified and evaluated to determine their adequacy to assure a quality product in keeping with the required safety function.

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, 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 with minimum probability of rapidly propagating fracture.

GDC 35 requires that an emergency core cooling system (ECCS) with abundant capacity be provided. 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 clean-up systems be provided to control fission products, hydrogen, oxygen, and other substances that may be released into the reactor containment. The staff reviewed the ESF structural materials to ensure that the requirements of GDC 41 have been met with respect to corrosion rates as they relate to hydrogen generation to post-accident conditions.

Appendix B, "Quality Assurance Criteria for Nuclear Power Plants," 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. The staff reviewed the ESF structural materials to ensure that the requirements of Appendix B have been 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.

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 ESF identified in Chapter 6 of the AP1000 design control document (DCD), consists of the containment vessel the passive containment cooling system (PCS), the containment isolation system, the passive core cooling system, the main control room emergency habitability system, and the fission product system. DCD Tier 2 Section 6.1 provides the AP1000 design requirements for ESFs materials.

In DCD Tier 2 Table 6.1-1, “Engineered Safety Features Pressure-Retaining Materials,” the material specifications for the principal pressure-retaining components are listed. The specifications for the core makeup tank, passive residual heat removal heat exchanger and valves in contact with borated water are listed in Table 5.2-1, “Reactor Coolant Pressure Boundary Materials Specifications.”

The materials for use in the ESF are selected for their compatibility with the reactor coolant system 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 specification 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 alloys in the ESFs in contact with borated water is limited to Alloy 690. Alloy 600 is used in limited areas for welding and buttering not in contact with the reactor coolant.

Low or zero cobalt alloy hardfacing materials in contact with the reactor coolant are qualified by wear and corrosion tests for equivalent performance to Stellite-6. The use of cobalt base alloys is minimized. Low or zero cobalt alloys used for hardfacing or other applications where 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, material specification for the particular type or grade of alloy. Austenitic stainless steel materials used in the ESFs components are handled, protected, stored, and cleaned to minimize contamination that could lead to stress corrosion cracking. 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 main control room emergency habitability system is SA-372 material and it is tested for Charpy-V notch energy per supplement S3 of material specification SA-372. The material is required 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 material is not permitted to be weld repaired.

The majority of the ESFs 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.

<u>Evaluation</u>

The staff evaluation of the ESFs materials is divided into the following four sections: materials and fabrication, composition and compatibility of ESFfluids, component and systems cleaning, and thermal insulation.

Materials and Fabrication

The staff reviewed DCD Tier 2 Section 6.1.1, “Metallic Materials” for ESF to determine the suitability of the materials for this application.

The components of the ESF 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. Protective coatings are reviewed in Section 6.1.2 of this report. 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 ESF will be limited to Alloy 690. Alloy 600 may be used for welding and buttering. The staff requested in request for additional information (RAI) 252.008 that the applicant provide information on what situations Alloy 600 would be used for welding and buttering and would it be in contact with the primary water. In its response dated November 8, 2002, the applicant responded that Alloy 600 is intended to be used for non-pressure boundary welding and buttering. If the Alloy 600 weld material is used for pressure boundary welding or buttering which will be in contact with the primary water, the last layer of weld material, as a minimum, will be Alloy 690. In addition, Revision 3 of DCD Tier 2 Section 6.1.1.1 was modified to include a statement addressing that Alloy 600 will not be in contact with the reactor coolant. The decision to use Alloy 690 was on the basis of its improved performance in pressurized water reactor (PWR) primary water. The staff believes the selection of Alloy 690 as the preferred nickel-based alloy is prudent because of its demonstrated improved resistance to stress corrosion cracking when compared with Alloy 600 which has been widely used for these applications in current 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 ESF, 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. This information is evaluated in Section 5.2.3 of this report.

The materials of ESF components comply with Subarticles NB-, NC-, and ND-3120 which require, in part, 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). Section 6.1.1.2 of the DCD 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.” Tier 2 Section 6.1.1.2 of the DCD 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 ESFs components are the same as those for ASME Code Class 1 components described in DCD Tier 2 Section 5.2.3.4. The staff review of this section and the staff review of conformance with the RGs noted above is documented in section 5.2.3 of this report.

The materials selected for ESFs satisfy the applicable requirements of Section III of the ASME Code and Section II of the Code. Therefore, 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 since 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. Thus, the design of the AP1000 ESFs meets the requirements of GDCs 1, 14, and 31, and 10 CFR 50.55a, with regard to ensuring an extremely low probability of leakage, rapidly propagating failure, or gross rupture.

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. Protective coatings are reviewed in Section 6.1.2 of this report.

Austenitic stainless steel plate conforms to ASME SA-240 and is confined to those areas or components which are not subject to a post-weld 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 post-accident 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 stress corrosion cracking. In the design of the AP1000, the potential for this is minimized by the release of trisodium phosphate 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 branch technical position (BTP) MTEB-6.1, "pH for Emergency Coolant Water for PWRs," regarding protection of austenitic stainless steel from stress corrosion cracking. Thus, the design meets the requirements of GDC 14 for ensuring the low probability of abnormal leakage or failure of the RCPB boundary and safety-related structures.

The AP1000 DCD Tier 2 Section 6.1.1 indicates that DCD Tier 2 Section 6.2.5 contains the hydrogen production analysis for a post accident analysis. However, this statement is incorrect since the AP1000 DCD does not contain a hydrogen generation analysis in anticipation of NRC completion of a rule change that would eliminate the design-basis hydrogen accident. Since this is not consistent with the current rule, the staff is not able to complete a review of the corrosion rates and consequent hydrogen generation. Therefore, this is draft safety evaluation report (DSER) Open Item 6.1.1-1. Additional discussion related to this issue is contained in Section 6.2.5 of this report.

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

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 assurance provided by performance of these or similar materials in current nuclear power plants (NPP) 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 ESF have demonstrated satisfactory performance in operating NPP 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 post-accident conditions, the ESF systems should not be susceptible to stress corrosion cracking 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 GDCs 4 and 35, and of 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 AP1000 design conforms with RG 1.37, “Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water Cooled Nuclear Power Plants” with an exception to quality standard American National Standards Institute (ANSI) N.45.2.1-1973 referenced in RG 1.37. The staff evaluation of quality assurance documents is found in Section 17.3, “Quality Assurance During Design, Procurement, Fabrication, Inspection and/or Testing of Nuclear Plant Items,” of this report. The staff finds the provisions for component and systems cleaning acceptable since these provisions conform with the regulatory positions of RG 1.37, with the exception evaluated in Section 17.3 of this report, and thus satisfy the quality assurance requirements of 10 CFR Part 50 Appendix B.

Thermal Insulation

The thermal insulation used in the AP1000 containment will be predominantly of the reflective metallic type. 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 ESF will be in conformance with RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels," 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 stress corrosion cracking resulting from the presence of contaminants in the thermal insulation. Therefore, the staff finds the thermal insulation used in the AP1000 design of the ESFs is acceptable since it conforms with the regulatory positions in RG 1.36. Compliance with the recommendations of RG 1.36 form a basis, in part, for meeting the requirements of GDCs 1, 14, and 31, and Appendix B to 10 CFR Part 50 with respect to ensuring that the reactor coolant boundary and associated auxiliary systems will have an extremely low probability of leakage, rapidly propagating failures, or gross rupture.

Conclusion

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 GDCs 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 pending resolution of DSER Open Item 6.1.1-1.

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 positions 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 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 Designed-Basis Accident (DBA) Conditions," and ASTM D 5144-00, "Standard Guide for Use of Protective Coating Standards in Nuclear Power Plants."

Summary of Technical Information

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

- inside containment,
- exterior surfaces of the containment vessel,
- radiologically controlled areas outside containment, and
- remainder of plant.

In addition, the AP1000 design addresses the classification of the coatings applied inside and outside containment in Table 6.1-2, "AP1000 Coated Surfaces, Containment Shell and Surfaces Inside Containment," based on their functions and to what extent their 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 coatings in various spaces of a NPP. DCD Tier 2 Appendix 1A through Revision 3 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 10 CFR Part 50, Appendix B, Quality Assurance requirements. 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 combined license (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

Revision 1 to RG 1.54, 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 RAI 281.001, requested the applicant to address how the AP1000 design incorporates Revision 1 to RG 1.54, “Service Level I, II, and III Protective Coatings Applied to Nuclear Power Plants,” July 2000, since 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, “AP1000 Coated Surfaces, Containment Shell and Surfaces Inside Containment,” meet the definitions of Service Levels I, II, and III. In its response dated December 2, 2002, the applicant stated that Section 6.1, “Engineered Safety Features Materials,” 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 Section 6.1, including Table 6.1-2, and determined that, with one exception, the applicant modified this section appropriately by incorporating the new guidance in the latest revision to RG 1.54. The applicant has committed to meet the guidance in RG 1.54, Revision 1, as appropriate for each of the following three service levels:

- Service Level I coatings are used in areas inside the reactor containment where the coating failure could adversely affect the operation of post-accident fluid systems and thereby impair safety.

- 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.

- Service Level III coatings are used in areas outside the reactor containment where failure could adversely affect the safety function of a safety-related structure, system, or component.

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

Conclusion

The staff concludes that the protective coatings system 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 the 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 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 serves both to prevent the uncontrolled release of radioactivity to the environment and to act as the passive safety-grade interface to the ultimate heat sink.

The primary containment has a design leakage rate of 0.10 weight percent (w/o) 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 full double-ended rupture of the main steamline (a 0.13 m^2 (1.4 ft^2) nominal area break based on the flow restrictor) from 30 percent power with a main steam isolation valve (MSIV) failure, and is 496.39 kPa (57.3 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 air flow 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, reactor coolant pumps, pressurizer, and associated connecting piping), the in-containment refueling water storage tank (IRWST), the core makeup tanks (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. There is no structural connection 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 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 with 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, MWt	1940	3400

Parameter	AP600	AP1000
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	(1.73E+06 ft^3) $4.9x10^4$ m^3	(2.06E+06 ft^3) $5.8x10^4$ m^3
Volume-to power ratio	(892 ft^3/MW) 25.2 m^3/MW	(606 ft^3/MW) 17.2 m^3/MW
Internal design pressure	(45 psig) 411.6 kPa	(59 psig) 508.12 kPa
External design pressure	(3.0 psid) 20.7 kPa	(2.9 psid) 20 kPa
Design temperature	(280 $^\circ$F) 138 $^\circ$C	(300 $^\circ$F) 149 $^\circ$C
Design leak rate, weight %/day	0.10	0.10
Calculated peak internal pressure (design margin)	(44.1 psig) 405.4 kPa (1.5 %)	(57.3 psig) 496.39 kPa (2.3%)
Calculated peak external pressure (design margin)	(2.0 psid) 13.8 kPa (33 %)	(2.4 psid) 16.6 kPa (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
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

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," is described in Section 15.3 of this report. 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.

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

The reliance of the AP1000 on cooling by naturally occurring physical phenomena represents a significant difference from other Westinghouse designs. 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 non-safety 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. The staff's evaluation of the combustible gas control is contained in Section 6.2.5 of this report.

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 powered from safety-grade alternating current (ac). The staff's evaluation of the containment isolation system is contained in Section 6.2.4 of this report.

<u>Compliance with Regulatory Requirements</u>

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 6.1.1.2.A, "PWR Dry Containments, Including Subatmospheric Containments." Westinghouse uses the <u>W</u>GOTHIC 4.2 computer program to evaluate the containment performance. The review of <u>W</u>GOTHIC 4.2 and the evaluation model used to evaluate containment performance can be found in Chapter 21 of this report.

<u>Compliance with 10 CFR Part 50, Appendix A</u>

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," is delineated in Chapter 6.2 of the SRP. 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. 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 (GDCs 16 and 50)

Acceptance criteria for existing containments include a margin between the design pressure and a conservatively calculated peak accident pressure. The margin varies 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 where much data and information are known, and the staff possessed an independent, confirmatory calculational capability, a 10 percent margin was expected at the CP stage to cover uncertainties in meeting GDCs 16 and 50 following final construction, 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 has 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.

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, 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. The staff review is documented in Chapter 21 of this report.

The staff has determined that the WGOTHIC 4.2 computer program, combined with the conservatively biased evaluation model described in NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design Docket No. 52-003," is acceptable for the evaluation of the peak containment pressure following a design basis accident for the AP1000 design as discussed in Chapter 21 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 conservative values which bound the range of most inputs, and applies conservative multipliers on the correlations used for PCS heat and mass transfer. Conservative models are used in the AP1000 <u>W</u>GOTHIC 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 LOCA, and up to about 1000 seconds for 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.

<u>Long Term Pressure Analysis (GDC 38)</u>

The objective of the long-term pressure analysis is to demonstrate that the containment design conforms to the objectives of 10 CFR Part 50 Appendix A, Criterion 38.

In Item II.b of Section 6.2.1.1.A of the SRP, the staff guidance used to evaluate compliance with GDC 38, "Containment Heat Removal," is 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. In current operating reactors, credit for this 50 percent reduction in pressure is considered in the siting evaluation. Westinghouse does not credit any leakage reduction due to decreased pressure. The AP1000 siting evaluation is performed with a constant, design basis leak rate. Westinghouse had originally 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 have been near the design value, and there was no need to demonstrate a pressure reduction for the leak rate assumption used in the siting evaluation.

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 (<u>W</u>GOTHIC). 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 performs an ancillary calculation to credit more PCS water in the evaporation process, effectively generating a correction factor, and applies it to the limited PCS flow model.

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. In addition, the calculated pressure is not used to demonstrate compliance with other regulatory requirements. Whether or not credit is taken for 2-D heat conduction, the staff finds the design to be in compliance with GDC 38 and the containment pressure and temperature following the limiting LOCA are maintained at acceptably low levels. 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.

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 staff therefore considers the 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.

Compliance with 10 CFR 52.47(b)(2)

The unique characteristics of PCS are explicitly recognized in the regulations governing the evaluation of standard plant designs. 10 CFR 52.47(b)(2)(i)(A) states that, 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. Since 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 do 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 21.3.4.4 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, "WGOTHIC Application to AP600 and AP1000," April 2002, must be used. Further, the assumptions must be consistent with the limitations and restrictions denoted in Section 21.3.4.4 of this report.

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 Westinghouse-GOTHIC (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 of January 15, 1997, on 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 (for up to 7 days). In addition, the PCCWST also provides makeup to the spent fuel pool and for fire protection.

The initial conditions of pressure, temperature, humidity, and net containment free volume used for DBA analyses are provided in Table 6.2-2.

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 %	
Net Free Volume	58,333 m^3	2.06E+06 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 TS maximums and have been shown, in Section 5 of WCAP-15846, "WGOTHIC Application to AP600 and AP1000," April 2002, to result in a conservative peak pressure calculation. It was also shown that 0 percent relative humidity is a conservative assumption. The staff has reviewed these input assumptions and finds them acceptable since they maximize the calculated peak containment pressure, consistent with the guidance in SRP 6.2.1.1.A.

The PCS flow rates and surface area coverage used for DBA analyses are provided in Table 6.2-3.

Table 6.2-3 PCS Flow Rates and Area Coverage

PCCWST Water Elevation		**Minimum Flow Rate**		**Area coverage**
ft	m	liters/min	gpm	% of circumference
27.4	8.35	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.

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 is 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 the reduced PCS water delivery rates as shown in Table 6.2-3. 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 Westinghouse's analyses of the AP1000 containment's pressure response, 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 for 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 for a standard Westinghouse two- or three-loop plant. Blowdown is the time during which the coolant system contents 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 by 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 comes into play during this first portion of a postulated transient. In Section 8 of WCAP-15846, 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 current operating reactors). This analysis showed that the multinode model during blowdown yields comparative results 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 via radiative, convective, and evaporative cooling from the containment outer surface.

For the LOCA events, two limiting double-ended guillotine reactor coolant system (RCS) pipe breaks are analyzed. In one case, the break is postulated to occur in the hot-leg of the RCS, and 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 post-blowdown peak pressure. The cold-leg break analysis includes the long-term contribution to containment pressure from the sources of stored energy, such as the steam generators. The LOCA mass and energy release calculations are discussed in Section 6.2.1.3 of this report.

For the MSLB event, a representative pipe break spectrum is analyzed. Various break sizes, power levels, and failure assumptions are analyzed with the WGOTHIC code. The MSLB mass and energy release calculations are discussed in Section 6.2.1.4 of this report.

A summary of the calculated pressures and temperatures for LOCA and MSLB postulated accidents are provided in Table 6.2-4.

Table 6.2-4 Summary of Calculated Pressures and Temperatures for LOCA and 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	417.11 (45.8)	91.0 (13.2) 17.9%	194.4 (381.9)	- - -

Break	Peak Pressure [kPa (psig)]	Available Margin[1] [kPa (psig)]	Peak Temperature[2] [°C (°F)]	Pressure at 24 hours [kPa (psig)]
LOCA, double-ended, cold-leg guillotine	483.29 (55.4)	24.8 (3.6) 4.9%	138.9 (282)	266.8 (24)
MSLB, 1.4 ft^2, full DER, 101% power, MSIV failure	471.57 (53.7)	36.5 (5.3) 7.2%	190.7 (375.3)	- - -
MSLB, 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)

The maximum calculated pressure in the primary containment occurs from an MSLB at 30 percent power. The maximum calculated pressure is 469.39 kPa (57.3 psig) at about 810 seconds after the MSLB begins. This value provides a margin of 2.3 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 Westinghouse identified conservatisms in 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 technical specification (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. A zero percent humidity initial condition is used to increase the initial stored energy inside containment.

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

- The PCS liquid film flow 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 1,666 liters/min (about 440 gpm). The higher initial flow rate for the AP1000, 1,775.7 liters/min (about 469.1 gpm), helps to offset the increase 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, Section 7 of WCAP-15846.

- The water coverage is obtained from the limiting flow model, as described in Section 7 of WCAP-15846, based on the wetted surface areas listed in Table 6.2-3. 2-D conduction is 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 when 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, upwards facing surfaces that may become covered with a condensation film is not credited.

The limiting LOCA and MSLB peak containment pressure and temperature calculations provided in support of the design certification are summarized in Table 6.2-4.

The staff has allowed Westinghouse to move the heat sink information from the 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, which is considered to be fully proprietary to Westinghouse Electric Company.

<u>Summary of Staff CONTAIN Analyses</u>

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, ADAMS Accession Nos. ML030850615, ML030860392 and ML030860396). These analyses indicate similar characteristics for the PCS performance for 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 objectives of 10 CFR Part 50 Appendix A, Criterion 38.

In Item II.b of Section 6.2.1.1.A of the SRP, the staff guidance used to satisfy GDC 38, "Containment Heat Removal," 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. In current operating reactors, credit for this 50 percent reduction in pressure is considered in the siting evaluation. Westinghouse does not credit any leakage reduction caused by decreased pressure. The AP1000 siting evaluation is performed with a constant, design-basis leak rate. Westinghouse had originally proposed that the calculated pressure reduction be based on 50 percent of the design pressure to meet the intent of GDC 38 and to be consistent with current operating reactors. The staff found this approach acceptable because the peak calculated pressures are near the design value, and there is no need to demonstrate the leak rate assumption used for the siting evaluation.

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.

The response for the limiting hot-leg break LOCA is provided in DCD Tier 2 Figure 6.2.1.1-9. 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. The response is shown in DCD Tier 2 Figure 6.2.1.1-1. This limiting case yields a peak containment pressure of 508 kPa (59 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. 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 pressure response is consistent with GDC 38 and the containment pressure following the limiting LOCA is maintained at an acceptably low level.

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. In addition, the calculated pressure is not used to demonstrate compliance with other regulatory requirements. Whether or not credit is taken for 2-D heat conduction, the staff finds the design to be in compliance with GDC 38 and the containment pressure and temperature following the limiting LOCA are maintained at acceptably low levels. 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 Westinghouse's maximum external pressure analysis satisfies the following requirement:

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

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. A -40 °C (-40 °F) ambient temperature and a steady 21.5 m/sec (48 mph) wind outside of containment were also assumed, to maximize the cooling of the containment atmosphere and maximize 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 negative 0.2 psig) was assumed, consistent with the TS 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 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 were found by Westinghouse not to be bounding.

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 a staff requirements memorandum, 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. The containment spray system is described in DCD Tier 2 Section 6.5.2 and the staff's evaluation of the system is found in Section 19.2.3.3.9 of this report.

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 main control room (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. 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 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, providing assurance that containment design conditions important to safety are not exceeded for the duration of accident conditions. The staff therefore finds Westinghouse's 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 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 6.2.1.3, "Mass and Energy Release Analysis For Postulated Loss-of-Coolant Accidents," forms the basis for concluding whether Westinghouse's subcompartment analysis satisfies the following requirements:

- GDC 4, regarding the appropriate protection of structures, systems, and components 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.

Selection of Postulated Breaks and Subcompartments

As discussed in DCD Tier 2 Section 6.2.1.2, Westinghouse has applied the leak-before-break (LBB) concept to the RCS high energy piping. LBB is applicable to RCS piping 6 inches 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 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. The LBB evaluation is provided in DCD Tier 2 Section 3.6, "Protection Against the Dynamic Effects Associated with the Postulated Rupture of Piping."

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. The staff's evaluation and acceptance of LBB for the AP1000 is discussed in Section 3.6.3 of this report.

Table 6.2-5 summarizes the postulated breaks and design pressures for the subcompartments analyzed. For all subcompartments, the postulated breaks envelope 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)
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 (ft^2) main steamline rupture	34.5 kPa (5 psid)

An evaluation was performed by Westinghouse 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 five-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 applied 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 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. The staff’s review and acceptability of this analysis is discussed in Section 6.2.8 of this report.

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) steam generator (SG) blowdown line break. For the 4 in pressurizer spray line break the Fauske break flow 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 results in a higher pressure at the break and thus a greater mass release. Since 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), WCAP-8078 (Non-Proprietary)), 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 6.2.1.2. In particular, this guidance is as follows:

- 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 6.2.1.2, for the CP stage of a review. However, the calculations show margins still exists. 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 6.2.1.3 regarding the mass and energy release used in the analysis by the assumption of a constant mass blowdown rate and use of 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 (Shepard, R. M., et al., "Westinghouse Mass and Energy Release Data for Containment Design," WCAP-8264-P-A, June 1975 (Proprietary), and WCAP-8312-A, Revision 2, August 1975 (Non-Proprietary)). 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," WCAP-10325, May 1983 (Proprietary)), and NOTRUMP has been found acceptable for use in currently licensed plants for small line breaks as discussed in Section 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 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, and the calculated differential pressures resulting from pipe breaks postulated in accordance GDC 4. Therefore, the staff finds Westinghouse's 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 was 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," in Section 6.2.1.2, "Subcompartment Analysis," 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 chapter.

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 (2250 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 break flow 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 steam generators.

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, “Mass and Energy Release Analysis for Loss-of-Coolant Accidents,” 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 (Westinghouse LOCA Mass and Energy Release Model for Containment Design, WCAP--10325-P-A (Proprietary) and WCAP-10326-A (Non-proprietary), 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. 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 6.2.1.4, and models the AP1000 passive safety features in a conservative manner. The post blowdown 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 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 1,500 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
- steam generator 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 dead band
- margin in RCS volume (+1.4percent)
- 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 steam generator stored energies are released at a conservatively high rate
- allowance in core stored energy for effect of fuel densification
- margin in core stored energy (+15.0 percent)
- allowance in initial pressure to account for instrument error and dead band
- margin in steam generator mass inventory (+10.0 percent)
- 1 percent of the Zirconium around the fuel is assumed to react

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, which increase the stored energy in the primary system to be consistent with the guidance in SRP 6.2.1.4 and 6.2.1.1.A to maximize the calculated containment pressure and temperature, to be acceptable for the licensing analyses.

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, "Westinghouse LOCA Mass and Energy Release Model for Containment Design - March 1979 Version," dated May 1983.

Description of Post-Blowdown 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 break flow 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 SGs 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 break flow 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 current operating Westinghouse PWRs are the ESFs. The safety features of current 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 time frame and are not included in the SATAN-VI model. CMTs cannot inject into the common direct vessel injection 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 RCSs 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 direct vessel injection (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 current 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 break flow is determined from a comparison of the total energy available release rate and the energy release rate assuming that the break flow was 100 percent saturated steam. Saturated steam releases maximize the calculated containment pressurization.

The staff reviewed the post-blowdown model, as it applies to the AP1000. The staff found the post-blowdown model, which increases the mass and energy released from the primary system to be consistent with the guidance in SRP 6.2.1.4 and 6.2.1.1.A to maximize the calculated containment pressure and temperature, to be acceptable for the licensing analyses.

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 singlefailure. 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. The staff's review of the initial conditions for LOCA analyses, the WGOTHIC code, and its results are discussed in Section 6.2.1.1 of this report.

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. Since 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. They have 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, 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 reactor coolant pumps, 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 passive residual heat removal (PRHR) heat exchanger. 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 via the PRHR heat exchanger. This causes a reduction of the heat transfer from the primary system to the SG secondary system and causes a reduction of mass and energy releases via the break.

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

Initial analyses, which considered single active failure of either one main steamline isolation valve or one feedwater isolation valve, determined that the main feedwater isolation valve failure is 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 completed, there is no additional mass or energy released to 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 case yields a peak containment pressure of 496.42 kPa (57.3 psig) at about 810 seconds into the event.

Significant Parameters Affecting Steamline Break Mass and Energy Releases

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 since they maximize the calculated peak containment pressure, consistent with the guidance in SRP 6.2.1.1.A.

Description of Blowdown Model and Mass and Energy Release Data

In this 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 (Carlin, E. L. and U. Bachrach, "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 since they are consistent with the guidance provided in SRP 6.2.1.4, “Mass and Energy Release Analysis for Postulated Secondary System Pipe Rupture,” Section II acceptance criteria.

Containment Response Analysis and Initial Conditions

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

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies of 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 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 Westinghouse's minimum containment pressure analysis satisfies the following requirements:

- 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, in part, 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, and 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.

The calculated containment backpressure used by Westinghouse for the AP1000 ECCS analysis is presented graphically in DCD Tier 2 Figure 6.2.1.5-1. 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
- a temperature of -18 °C (0 °F) was assumed in the shield building annulus
- 10 percent was added to the containment volume
- passive heat sink surface areas were increased by a factor of 2.1
- during the blowdown period inside containment, the Tagami heat transfer correlation with a multiplier of 4 was used
- for the post-blowdown period inside containment, the Uchida heat transfer correlation with a multiplier of 1.2 was used
- containment purge was assumed to be in operation through two 38.1 cm (15 in) diameter lines (16-inch 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 CSB 6-1, "Minimum Containment Pressure Model For PWR ECCS Performance Evaluation" of SRP 6.2.1.5. The mass and energy releases used in the minimum containment pressure analysis were determined by the requirements of Appendix K to 10 CFR Part 50, and are described in WCAP-14171-P (WCOBRA/TRAC). These mass and energy releases are consistent with SRP 6.2.1.5, which specifies that the releases should be on the basis of 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 there is no line for it to spill from.

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. Westinghouse's minimum backpressure analysis does not assume the containment recirculation cooling system to be operating. At about six seconds following the initiation of the accident, the containment recirculation cooling system would be secured on a containment isolation signal, and that the impact of operation of the cooling system for six seconds would be small. Because the break flow 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 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 that a conservative backpressure be used in ECCS reflood calculations, and has 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 6.2.1.5. Furthermore, Westinghouse has followed the guidance given in SRP 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 acceptable. The acceptability of the credited backpressure has been evaluated in the overall context of the ECCS performance capability studies. The staff's evaluation of the ECCS performance is provided in Section 15.2.6.5 of this report.

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. Testing and inservice inspection of the containment vessel is described in DCD Tier 2 Section 3.8.2.6, while isolation testing is described in DCD Tier 2 Section 6.2.3 and leak testing is described in DCD Tier 2 Section 6.2.5. The valves of the PCS are periodically stroke tested, and DCD Tier 2 Section 6.2.2 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. See DCD Tier 2 Section 3.8.2 for the requirements for inservice inspection of the steel containment vessel. DCD Tier 2 Section 6.2.2 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. The collection of condensate from the containment shell and its use in leakage detection are discussed in DCD Tier 2 Section 5.2.5.

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

<u>Preoperational Testing</u>

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, 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 (DCD Tier 2 Table 6.2.2-1):

1775.7 L/m (469.1 gpm) at the minimum operating water level
857.8 L/m (226.6 gpm) at a level after the first standpipe is uncovered
667.4 L/m (176.3 gpm) at a level after the second standpipe is uncovered
545.9 L/m (144.2 gpm) at the level after the third standpipe is uncover

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 re-established 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 and 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-3 in) 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 air flow 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 passive containment cooling water storage tank will be in excess of 381.12 L/m (100.7 gpm). Temporary instrumentation or changes in the passive containment cooling water storage tank 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/m (100.7) gpm for a duration of 4 days.

The passive containment cooling water storage tank 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/m (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).

Additional details for preoperational testing of the PCS are provided in DCD Tier 2 Chapter 14, and discussed in Section 14 of this report.

The staff finds 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. The initial test program is described in DCD Tier 2 Section 14.2.9.1.4, “PCS.”

<u>Operational Testing</u>

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, 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 water flow delivery, consistent with the accident analysis.

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

- Test frequency is consistent with the plant TSs (DCD Tier 2 Section 16.3.6) 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 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.
10 CFR 50.34(f)(2)(xvii) requires instrumentation 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)
- noble gas effluents at all potential accident release points

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

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 the ESF actuation system operation is described in DCD Tier 2 Section 7.3.

The containment hydrogen concentration is measured by the hydrogen concentration monitoring subsystem. The system is described in DCD Tier 2 Sections 6.2.4 and 7.5 and was evaluated by the staff in Section 6.2.5 of this report. The response time of the sensor is at least 90 percent in 10 seconds, as indicated in DCD Tier 2 Table 6.2.4-1. As part of the preoperational and inservice testing programs, the COL applicant is responsible for verifying that the response time of the procured instrument meets the recommendations of Item II.F.1 of NUREG-0737, “Clarification of TMI Action Plan Requirements.” As detailed in DCD Tier 2 Section 1.9, the hydrogen monitoring system is designed in compliance with the recommendations of NUREG-0737.

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 72-foot elevation to the 110 foot elevation. The staff concluded that containment cooling status can be determined through alternative means to direct reading of containment sump water temperature. The alternative means include either Category 2 residual heat removal heat exchanger 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 heat exchanger 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

Information concerning the operation of the AP1000 passive core cooling system, which includes a description of the design features of the system’s debris screens, is provided in DCD Tier 2 Section 6.3. The AP1000 has two sets of screens, which are the IRWST screens and the containment recirculation screens. A description of these screens, their design criteria, and their conformance with Revision 2 of RG 1.82 is included in DCD Tier 2 Section 6.3.2.2.7. As discussed further in Section 20 of this report, the staff reviewed the AP1000 debris screens in accordance with the current state of knowledge concerning the issues associated with Generic Safety Issue 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 post-accident debris in many reactor designs is the thermal insulation on piping and components of the RCS and other associated and co-located systems. In order to limit the challenge to the IRWST and containment recirculation screens from insulation debris, the applicant has 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.

The zones considered by the applicant to be vulnerable to damage by jet impingement are defined in DCD Tier 2 Section 6.3.2.2.7.1 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, are 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. As 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 staff's SER on the BWROG's report NEDO-32686 states that a spherical impingement model appears logical for congested zones of containment, and it may be the best approximation for estimating the amount of debris in congested zones. However the staff's 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 agrees 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 is 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, due to 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, the staff concludes that the applicant has 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 is DSER Open Item 6.2.1.8.1-1.

In zones vulnerable to jet impingement, the DCD states that reflective metallic insulation (RMI) will be used, or an equivalent material that will not be damaged by jet impingement or that will not transport to the containment recirculation screens. Testing sponsored by the NRC and the BWROG in the resolution of the 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 with 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. As 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 and 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 post-accident 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. As 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 to be closed.

Similar to its position concerning the AP600, the applicant maintains 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 10 CFR Part 50, Appendix B. 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 the staff 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 of the adequacy of the IRWST and containment recirculation screens.

DCD Tier 2 Section 6.3.8.1 states that 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." As 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 passive core cooling system.

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 staff's assessment of the ability of the IRWST and containment recirculation screens to accommodate anticipated quantities of post-accident debris is provided in the two subsequent sections.

6.2.1.8.2 Pool Transport and Head Loss Evaluation of the IRWST Screens

The IRWST screens are described in DCD Tier 2 Section 6.3.2.2.7.2 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. 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 ADS spargers' being located 4.87 m (16 ft) above the IRWST floor and only on one side of the tank, and the sequencing of the ADS valves. As the applicant did not provide a quantitative analysis of the turbulence conditions within the IRWST during an ADS actuation, however, the staff lacks a sound basis to conclude that 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.

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, or 500 lb) was consistent with estimates made for current generation PWRs in the Generic Safety Issue (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 was 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 considers the capability of the AP1000 IRWST screens to accommodate anticipated debris loadings to be DSER Open Item 6.2.1.8.2-1.

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

The containment recirculation screens are described in DCD Tier 2 Section 6.3.2.2.7.3 as being flat, vertical screens, each 6.5 m^2 (70 ft^2) in area. 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 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 five 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 post-accident 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 flood-up (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 considers debris blockage in the RCS to be DSER Open Item 6.2.1.8.3-1.

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, or 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 is DSER Open Item 6.2.1.8.3-2.

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 post-accident 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, or 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 was 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 considers the capability of the AP1000 containment recirculation screens to accommodate anticipated debris loadings to be DSER Open Item 6.2.1.8.3-3.

6.2.1.8.4 Conclusion

The staff completed its review of the adequacy of the IRWST and recirculation screens' performance in light of anticipated post-accident debris loadings. With the exception of the DSER open items associated with RAIs 650.001, 650.004, 650.005, and 650.006 that are documented in this evaluation, the staff finds that debris screen performance has been acceptably addressed for the AP1000 passive core cooling system.

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 post-accident 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 and 1.29. 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 three 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.

- design components necessary for accident mitigation to remain functional during, and 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, the PCS water storage tank (PCCWST), the air baffle, air inlets, and air diffuser, and the water distribution system comprising a water distribution bucket and distribution weirs. The design of the shield building is discussed in Section 6.2.3 of this report.

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), air-operated valve and a normally open, dc-powered, motor-operated valve. 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 2,857,990 liters (755,000 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 flow paths. In the event of a DBA, heat removed from the containment atmosphere through the vessel wall heats the air in the annular flow path 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 air flow 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 surround 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. The staff;s evaluation of the containment and shield building is provided in Section 3.8 of this report.

6.2.4 Containment Isolation System

The containment isolation system 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 containment isolation system is to permit the normal or post-accident passage of fluids through the containment boundary while protecting against release to the environment of fission products 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 containment isolation system. 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 piping systems penetrating containment.

The staff's review encompassed the following areas specified by Section 6.2.4 of the SRP:

- containment isolation system 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.
    - use of closed systems as isolation barrier substitutes for valves.
- the protection provided for containment isolation systems against loss of function due to missiles, pipe whip, and natural phenomena.

- environmental conditions in the vicinity of containment isolation systems 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.

The discussion of the staff's findings and conclusions for each of the above review areas is provided below.

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." 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 GDCs. 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 containment 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) is that all non-essential systems shall be automatically isolated upon initiation of an appropriate containment isolation signal. Non-essential 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 post-accident 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 two 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, non-automatic isolation valves have provisions for locking the valves in the closed position. Verification that non-automatic isolation valves are in the correct position during plant operation is through administrative controls and the design of locking devices.

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. 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 safe-shutdown earthquake, 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, 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. This requirement is included in the design bases for the AP1000 containment isolation system (DCD Tier 2 Section 6.2.3.5).

6.2.4.3 Normal and Fail Positions of Isolation Valves

The acceptance criteria in Section 6.2.4 of the SRP state 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 non-motor-operated automatic isolation devices fail in the closed position upon loss of power source (air or electrical power). Motor-operated valves 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. Compliance with this requirement is assured by the TSs.

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:

- containment isolation signal (DCD Tier 2 Section 7.3.1.2.1)

  A containment isolation signal 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

- safeguards actuation signal (DCD Tier 2 Section 7.3.1.1)

  A safeguards actuation signal is initiated by any one of the following monitored variables:

  - low pressurizer pressure
  - low lead-lag compensated steam line pressure
  - low reactor coolant inlet temperature
  - High-2 containment pressure
  - manual initiation

- steamline isolation signal (DCD Tier 2 Section 7.3.1.2.10)

  A steamline isolation signal is initiated by any of the following monitored parameters:

  - High-2 containment pressure
  - low reactor coolant inlet temperature
  - low lead-lag compensated steam line pressure
  - high steamline pressure negative rate
  - manual initiation

- main feedwater isolation signal (DCD Tier 2 Section 7.3.1.2.6)

  A main feedwater isolation signal is generated by any of the following monitored parameters:

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

- startup feedwater isolation signal (DCD Tier 2 Section 7.3.1.2.13)

  This signal occurs as the result the following conditions:

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

- SG blowdown isolation signal (DCD Tier 2 Section 7.3.1.2.11)

  A SG blowdown isolation signal is used for SG blowdown line isolation. This signal is initiated by either of the following parameters:

  – PRHR heat exchanger alignment signal
  – low narrow range SG water level

- Normal Residual Heat Removal (RNS) System Isolation (DCD Tier 2 Section 7.3.1.2.20)

  Automatic isolation of the RNS system containment isolation valve is initiated by:

  – High-2 containment radioactivity
  – Automatic or manual safeguards actuation signal. This signal 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 system to permit RNS system operation after a safeguards signal.

- containment air filtration system isolation signal (DCD Tier 2 Section 7.3.1.2.19)

  – Automatic or manual safeguards actuation signal
  – manual actuation of containment isolation
  – manual actuation of passive containment cooling
  – High-1 containment radioactivity

The following non-safety-grade signal is also used for automatic containment isolation:

- diverse actuation system (DAS) signal (DCD Tier 2 Section 7.7.1 11)

  The DAS is a non-safety-related instrumentation system that provides diverse backup to support risk goals.

RG 1.141 and TMI Item II.E.4.2 state that containment isolation system 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 a 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 has indicated a containment isolation actuation pressure of ≤ 8 psig for the AP1000 TSs. This setpoint was used in all applicable DBA analyses. However, a Reviewer's Note in the AP1000 TSs states that the 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 the TMI Item II.E.4.2 guidance.

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 the SRP 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 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 main steamline isolation valves, and have been included in the AP1000 design. For valve sizes up to 12 inches, 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

The staff's review of redundancy for valved piping penetrations is discussed under Section 6.2.4.1 above. The AP1000 containment design incorporates certain non-valved 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, these penetrations are also listed in DCD Tier 2 Table 6.2.3-1.

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 are 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 tubesheet and secondary system piping actually serve as a containment boundary. The SG piping penetrating containment (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 GDC 57 criteria, 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 containment isolation system design bases include protection from missiles, pipe breaks, earthquakes, fire, internal and external flooding, ice, wind, and tornados. Specific features and design criteria for protection of systems, structures, and equipment from these phenomena is discussed in other sections of this report.

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 containment isolation system has been properly classified to ensure that protection from these environmental hazards is encompassed by the isolation system mechanical and electrical design bases and quality standards. The staff's review of the environmental qualification of the AP1000 structures, systems, and components, including containment isolation equipment, is discussed in Section 3.11 of this report.

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

The containment isolation system 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 containment isolation system.

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. See Section 6.2.6 of this report for 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-inch) 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 assumptions and criteria for radiological dose analyses and ECCS analysis (reflood backpressure) assumptions use 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 inch) 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 very infrequently. As a result, the containment vent/purge system should only be used for containment pressure control, as low as reasonably achievable or air quality considerations for personnel entry, or for TS surveillances. This restriction is included in TS SR 3.6.3.1.

6.2.4.15 Conclusion

The staff has determined that the containment isolation system meets the acceptance criteria of Section 6.2.4 of the SRP, including the NUREG-0737 TMI requirements.

6.2.5 Containment Combustible Gas Control

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.

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.” The staff used the guidance, staff positions, and acceptance criteria of SRP Section 6.2.6 and RG 1.163 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 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 Containment Integrated Leakage Rate Type A Tests

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 post-repair 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 water-sealed 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 four hours before beginning the test.

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

<u>Test Method for Type A Tests</u>

The DCD indicates that, consistent with ANSI/ANS-56.8-1994, the "absolute" method and the “mass point” method will be used for Type A tests. 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 and the guidance of RG 1.163.

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.

<u>Test Acceptance Criteria</u>

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, SRP 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.

<u>Provisions for Additional Testing in the Event of Failure to Meet Acceptance Criteria</u>

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 Containment Leakage Rate Testing Program requirements.

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 four containment penetration types that will receive preoperational and periodic Type B tests are:

- penetrations having resilient seals, gaskets, or sealant compounds;
- air locks and associated door seals;
- maintenance and equipment hatches and associated seals; and
- electrical penetrations.

This includes one main equipment hatch, two personnel air locks, one fuel transfer tube, a maintenance hatch, 32 electrical penetration assemblies, and three 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 flowmeter makeup flowrate 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. The combined Types B and C acceptance criterion is 0.6 L_a. In addition, air lock chambers and individual doors must meet 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 will be tested with the containment (i.e., during Type A testing, the SG secondary side will be opened and vented to the atmosphere). The AP1000 is a PWR; therefore, these valves are not encompassed by Appendix J, Option A, paragraph II.H, which identifies those isolation valves for which Type C testing requirements are applicable, or by RG 1.163 and ANSI/ANS-56.8-1994, which provide similar guidance for Option B of Appendix J. 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 flowmeter 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

The staff reviewed the proposed TSs in DCD Tier 2 Chapter 16. The staff determined that the proposed TS are consistent with staff guidance for format and content of TSs for containment leakage rate testing, with one exception, discussed below. As stated above, each COL applicant is required to develop a "Containment Leakage Rate Testing Program." This program is a licensee-controlled document that is invoked by reference in the TSs.

The staff found one exception to staff guidance for format and content of TSs for containment leakage rate testing. The exception is that the numerical value of P_a should be stated in the TS, but it is not.

TS 5.5.8, “Containment Leakage Rate Testing Program,” includes the following passage:

- 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].

Option B of Appendix J to 10 CFR Part 50 requires the numerical value of P_a to be specified in the TS. The AP600 proposed TS provided the numerical value for P_a , correctly, but this has been changed for AP1000. [Note: 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 today 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. Also, the AP1000 proposed TS follow the Option B model.]

In RAI 480.010, the staff requested that Westinghouse provide justification for why the AP1000 differed from the AP600 in the treatment of this item. Westinghouse provided a response to this concern in a letter dated April 11, 2003 (ADAMS Accession No. ML031050025). However, the item remains open, as discussed below.

In a response to RAI 480.010, dated April 11, 2003, Westinghouse provided the following as justification for not placing the numerical value of P_a into the Technical Specifications:

1. It is simpler and reduces future changes to the DCD, and is consistent with the overall TS improvement strategy to minimize the need for a plant license amendment or Bases update for parameters that are expected to change due to re-analysis.

2. It is not clear that Appendix J specifically requires that the numerical value for P_a be included in the Technical Specifications. Appendix J, Option B, states that P_a is specified "...in the Technical Specifications." Westinghouse assumes that this is a reference to the entire Technical Specifications document, which includes the individual technical specifications and the associated bases.

3. The definition of P_a in Option B, "...the calculated peak containment internal pressure related to the design basis loss-of-coolant accident...", is incorrect for AP1000, since the limiting calculated peak containment internal pressure in DCD 6.2 occurs for a steamline break accident.

To resolve the issue, Westinghouse plans to revise the TS Bases to state the numerical value of P_a.

The staff carefully considered the requirements of Appendix J and the objectives of the TS improvement program when developing the latest revision of the Standard Technical Specifications. The staff determined that, despite the inconvenience for future plant-specific license amendments, Appendix J, Option B, requires the numerical value of P_a to be stated in the Technical Specifications, not in the Bases. This is reflected in the Standard Technical Specifications. Westinghouse's proposed resolution of this issue is therefore unacceptable.

Also, Westinghouse's assertion, that the definition of P_a in Option B is incorrect for AP1000, is in error. P_a is not meant to bound the calculated peak containment internal pressures of all postulated accidents. P_a is a parameter specifically established for the purpose of radiological consequence analysis and containment leakage rate testing. For this reason, only accidents that produce a significant radioactive source term in the containment are considered when the value of P_a is determined. Steamline breaks in the AP1000 do not produce a significant radioactive source term in the containment. Of course, containment design pressure must bound the calculated peak containment internal pressures of all postulated accidents, but containment design pressure is not the same as P_a. Thus, the design basis loss-of-coolant accident pressure is the correct parameter for determining the value of P_a. For the reasons stated above, this is DSER Open Item 6.2.6.4-1.

6.2.6.5 Conclusion

On the basis of its review, pending the resolution of DSER Open Item 6.2.6.4-1, 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 leak-tight 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.

6.2.7 Fracture Prevention of Containment Pressure Boundary

GDC 1 requires that SSCs important to safety shall 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 leak tight 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 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 that containment pressure boundary materials meets the requirements of GDC 1, 16, and 51.

Summary of Technical Information

The DCD Tier 2 Section 3.8.2 indicates that SA738, Grade B, material will be used for the AP1000 containment vessel. The 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 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 design and construction in accordance with the requirements of the ASME Code, Subsection NE-2000. However, the staff requested in its RAI 252.009 that the following requirements be provided to supplement the requirements of specification SA-738 and include these requirements in the AP1000 DCD:

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

- Supplementary Requirement S20, “Maximum Carbon Equivalent for Weldability,” of Material Specification SA-738 also applies to the material.

These two requirements are needed to ensure adequate material properties and weldability of the containment vessel material. The ASME Code, Section III, exempts SA-738, Grade B, material up to 1.75 inch of thickness from post-weld stress relief heat treatment. The AP1000 containment vessel is 1.75 inch thick. Because the containment vessel material thickness is 1.75 inch 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 non-metallic inclusions such as oxides and silicates will be minimized as a result of the vacuum-degassing of the steel. Requiring the S20 carbon equivalent weldability check will ensure that the steel is readily weldable.

Westinghouse responded to RAI 252.009 and in Revision 3 revised the DCD to require that supplementary requirements S1.7 and S20 be specified for the AP1000 containment vessel material.

<u>Conclusion</u>

Based on the review of the information included in the AP1000 DCD and the fact that the applicant will meet the requirements of 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 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 GDCs 1, 16, and 51 will be met.

6.2.8 In-Containment Refueling Water Storage Tank Hydrodynamic Loads

The IRWST is described in DCD Tier 2 Section 6.3.2.2.3 and Table 6.3-4. It is 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 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 hydrodynamc 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 steam flow.

For the AP600, the hydrodynamic loads were determined based on tests conducted at the VAPORE (Valve and Pressure Operating Related Experiments) test facility. The tests were divided into Phase A and Phase B. The Phase A tests are described in WCAP-13891, Revision 0, and the Phase B tests are described in WCAP-14324. Phase A tests simulated ADS operation through the submerged sparger in order to evaluate the hydraulic performance of the sparger under various steam flow 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 staff RAI Number 220.001, Westinghouse stated that the AP600 Automatic Depressurization System hydraulic tests were used to define loads on the AP1000 IRWST. Two tests were selected as representative of the sparger discharge pressures. One test simulates 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 spragers. The other test simulated a pressure time history representing the inadvertent opening of the second or third stage of ADS at full pressure. The latter test is characterized by pure steam flow.

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

Westinghouse states that the time histories from these two tests are applicable to the AP1000 since 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 and the fluid conditions are similar.

Since 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 passive core cooling system (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 passive residual heat removal heat exchanger (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, and 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 trisodium phosphate (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 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 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 whose failure 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 reactor coolant pressure boundary 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 reactor coolant pressure boundary are not exceeded.

- GDCs 35, 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 the 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 safety-related functional performance criteria of the PXS are described in the following sections.

6.3.1.1 Emergency Core Decay Heat Removal

For non-LOCA events, where a loss of core decay heat removal capability via the SGs occurs, the PRHR HX is designed to automatically actuate to remove core decay heat to prevent water relief through the pressurizer safety valves; cool the RCS to 215.6 °C (420 °F) within 36 hours, with or without reactor coolant pumps operating; continue decay heat removal operation for an indefinite time in a closed-loop mode of operation in conjunction with the PCS; and sufficiently reduce RCS temperature and pressure during a SG tube rupture event to terminate break flow, 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 makeup 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 (June 30, 1994 staff requirements memorandum (SRM) has 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 necessary reactivity control to maintain the core in a subcritical condition, and by providing residual heat removal capability to maintain adequate core cooling. The systems required for safe-shutdown are discussed in DCD Tier 2 Section 7.4.

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 reactor coolant pumps 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 on 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 are lost for 24 hours. Therefore, for LOCAs or other postulated events where 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 post-accident 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 stress corrosion cracking 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 an onsite electric power system and an 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 uninterruptible power supply 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. The piping and instrumentation drawings of the PXS are shown in DCD Tier 2 Figures 6.3-1 and 6.3-2. A summary of equipment parameters for the major components is contained in DCD Tier 2 Table 6.3-4.

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 107-foot 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 (2,500 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 3,400 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 motor-operated valve, 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 non-condensible gases that could interfere with CMT operation. The discharge line has two parallel, normally-closed, air-operated isolation valves that will open on a loss of air pressure or electric power, or on control signal actuation, to begin CMT injection. Downstream of the air-operated valves, 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 pre-operational 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.

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

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 cubic meters (2,000 cubic feet), are filled with borated water at a concentration of about 2,600 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 a motor-operated valve, a flow-tuning orifice, and two swing-disc check valves in series. The motor-operated valve 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 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 a motor-operated valve, 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 the 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 motor-operated valve (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 flow paths from the containment to the reactor.

The actuation signals and logic, as well as the permissives and interlocks, to align the IRWST injection and containment recirculation are described in DCD Section 7.3.1.2.2 and Table 7.3-1, and the actuation setpoints are specified in Table 3.3.2-1 of the AP1000 TS.

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. TS require visual inspections of the screens during every refueling outage to ensure they are not restricted by the debris. The design of the IRWST and recirculation screens and the design criteria are discussed in DCD Tier 2 Section 6.3.2.2.7. The staff evaluation of the IRWST and recirculation screens is discussed in Section 6.2.1.8 of this report.

6.3.2.4 pH Adjustment Baskets

The PXS utilizes pH adjustment baskets to control post-accident 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 post-accident floodup 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. The evaluation of the adequacy of the pH adjustment baskets is discussed in Section 15.3, "Radiological Consequences of Accidents," of this report under the heading of "Post-accident Containment Water Chemistry Management."

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 heat exchanger. 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 6.3-4. 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 (reactor coolant pump suction).

The PRHR HX performs emergency core decay heat removal for events not involving a loss of coolant. The heat exchanger is elevated above the RCS loops to induce natural circulation flow through the PRHR HX when the reactor coolant pumps are not available. The PRHR HX inlet line contains a normally open, MOV. This alignment maintains the heat exchanger full of reactor coolant at the RCS pressure. The outlet line contains two parallel, normally closed, air-operated valves that open on 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 flow path for the single-failure assumption of an inoperable valve in the safety analysis. The discharge valve opening time delays assumed in the safety analyses are provided in DCD Tier 2 Table 15.0-4b. The water temperature in the heat exchanger 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 heat exchanger with the reactor coolant pumps 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 heat exchanger 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.

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

6.3.2.6 Automatic Depressurization System

The ADS has a total of 20 valves divided into two identical groups, each consisting of four 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 a 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 stage 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. The ADS actuation logic is discussed in DCD Tier 2 Section 7.3.1.2.4 and summarized in Table 7.3-1.

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; and 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 set point 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 15.6.5-7 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 Low Differential Pressure Opening Check Valves

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 (June 30, 1994, SRM) to maintain current licensing practices for passive component failures in passive light-water reactor (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 flow paths 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 its safety functions.

Therefore, these check valves are considered active components subject to 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 back flow 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. There is only a low probability 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 back flow. 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 (1,550 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 failure-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.

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 non-condensible gas volume; accumulator level and pressure; IRWST level and temperature; and PRHR HX inlet line non-condensible 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 failure. 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 re-opening are the two exceptions. Single active failures are considered in Chapter 15 DBA analyses. 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 flow path. SECY-94-084 states the Commission-approved position that, consistent with current licensing practices, passive 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 NPP, and that only on a long-term basis does the staff consider passive 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 flow path to the core to supply sufficient flow to keep the core covered and to remove decay heat. The PXS flow paths 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 flow paths.

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 6.3-5 provides a summary of the failure mode and effect analysis of the PXS active components. Each of the valves in the PXS (including check valves, isolation valves, air- 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. Long-term cooling analysis results of various breaks presented in DCD Tier 2 Section 15.6.5.4C.3 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. The staff evaluation of this issue is included in Section 15.2.7 of this report on long-term cooling.

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 pre-operational inspection and testing, and in-service inspection and testing. Preoperational inspections are performed to verify that important elevations associated with the PXS components are consistent with the accident analyses. The pre-operational testing of the PXS is described in DCD Section 14.2.9.1.3. This testing includes valve inspection and testing, flow testing, and verification of heat removal capability.

Two basic types of in-service testing are performed on the PXS components: periodic exercise testing of active components during power operation, and operability testing of specific PXS features during plant shutdown. The PXS includes specific features to support in-service test performance. These include (1) remotely operated valves can be exercised during routine plant maintenance; (2) level, pressure, flow, and valve position instrumentation is provided for monitoring required PXS equipment during plant operation and testing; and (3) permanently installed test lines and connections are provided for operability testing. DCD Tier 2 Section 3.9.6.2 provides a description of the in-service 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. 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, whose failures could have an adverse effect on the PXS. There is not a non-safety-related system whose failure could reduce the functioning of the PXS. Therefore, the PXS meets position C.2 of RG 1.29, “Seismic Design Classification,” and 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, “Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants,” 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 stage 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 under the heading “Safety Classification of Passive Core Cooling System,” that states that, for systems that provide ECCs functions, full radiography in accordance with the requirements of ASME code, Section III, 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 control. 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 6.3-3 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. Remotely operated isolation valves, such as the isolation valves on the PRHR inlet line, the CMT cold-leg balance lines, and the accumulator and IRWST discharge lines, and the ADS fourth-stage MOVs, which are normally open and remain open during PXS operation, are required by the AP1000 TS to 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. The interlock features are discussed in DCD Tier 2 Section 7.6.2, “Availability of Engineered Safety Features,” and the staff evaluation is discussed in Section 7.6, “Interlock Systems Important to Safety,” of this report. 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 ICSB-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 have 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 post-accident 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. The instrumentation and controls are discussed in DCD Tier 2 Chapter 7, and the staff's review is discussed in Chapter 7 of this report.

DCD Tier 2 Section 6.3.7.4.1 provides a design description of the CMT level instrumentation using differential pressure instruments. The arrangement of the CMT differential pressure level instrument is shown in DCD Tier 2 Figure 6.3-1. Each CMT has ten level channels. Two wide-range level channels, which are not qualified for post-accident 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 post-accident 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. The load combinations, stress limits, and analytical methods for structural evaluation of the PXS for various plant conditions are discussed in DCD Tier 2 Section 3.9.3, and the requirements for dynamic testing and analysis are discussed in Section 3.9.2. Seismic design is discussed in DCD Tier 2 Sections 3.7, 3.8 and 3.10. Environmental qualification of equipment is discussed in DCD Tier 2 Section 3.11. Protection against missiles and from fire are discussed in DCD Tier 2 Sections 3.5 and 9.5.1, respectively. 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 Shutdown Events

During plant shutdown conditions, some of the PXS equipment is isolated to allow for maintenance of the system, and the normal residual heat removal system (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:

- loss of startup feedwater during hot standby, cooldowns, and heatups
- loss of RNS cooling with the RCS pressure boundary intact
- loss of RNS cooling during mid-loop operation
- loss of RNS cooling during refueling

In DCD Tier 2 Section 19E4, the applicant provides 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 engineered safety features on plant transient response such as DNBR, peak RCS pressure, and peak cladding temperature. The staff evaluation of shutdown operation is evaluated separately in Section 19.3 of this report. 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 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 via 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 the PXS system functional performance capability. An inadvertent opening of a 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 of a decrease in heat removal by the secondary system. A single SG tube rupture (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. These events were analyzed in DCD Tier 2 Sections 15.1, 15.2, and 15.6, respectively. A postulated double-ended rupture of one PRHR HX tube is not analyzed in DCD Tier 2 Section 15.6. 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 non-limiting and is covered by the effect of postulating a hot-leg or cold-leg break location considered in the break spectrum. The post-LOCA long-term cooling is analyzed in DCD Tier 2 Section 15.6.5.4C.

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 of 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 non-consequential 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 reactor coolant pump 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 the most reactive control rod stuck out of the core to comply 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 Chapter 15 analyses results 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 reactor coolant pressure boundary 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 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 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. The review of these codes, as well as the test programs, are discussed in Chapter 21 of this report.

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 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. There is only one potential need 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 via 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 one 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, “AP1000 Implementation of the Regulatory Treatment of Nonsafety-Related Systems Process,” dated November 2002, 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 provide makeup water to the PCS water storage tank to maintain external containment cooling water flow, and therefore provide the containment cooling and ultimate heat sink. The staff evaluation of this post-72-hour support action is described in Section 22.5.6 of this report.

6.3.5 Limits on System Parameters

The plant TS establish PXS operability requirements for reactor operation. The limiting conditions for operation and SR of various PXS subsystems are specified in TS 3.4.12 through 3.4.14, and 3.5.1 through 3.5.8. 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 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 are also provided in the TSs. The staff's evaluation of the TSs is addressed in Chapter 16 of this report.

6.3.6 Conclusion

The staff reviewed DCD Tier 2 Section 6.3 and other relevant material regarding the AP1000 passive core cooling system (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 and 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 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 reactor coolant pump trip and the results of the design-basis events are considered in the safety analyses to demonstrate meeting 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 pre-operational 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 the NUREG-0800, "Standard Review Plan," (SRP), 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, "Environmental and Dynamic Effects Design Bases," regarding accommodating the effects of, and being compatible with, postulated accidents, including the effects of the release of toxic gases.

- GDC 5, "Sharing of Structures, Systems, and Components," as related to shared systems and components important to safety.

- GDC 19, "Control Room," regarding maintaining the control room in a safe, habitable condition under accident conditions by providing adequate protection against radiation and toxic gases. 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 10 CFR Part 50.

- TMI-related requirement 10 CFR 50.34(f)(2)(xxviii), as it relates to evaluating potential pathways for radioactivity and radiation that may lead to control room habitability problems.

- 10 CFR Section 50.34(f)(2)(xxviii) (TMI Action Plan Item III.D.3.4 (NUREG-0737)) requirements, as they relate to providing protection against the effects of release of toxic substances, either on or off the site.

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 structures, systems, or components with other nuclear power units. Therefore, the air conditioning, heating, cooling and ventilation 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:

- the nuclear island non-radioactive (VBS) HVAC system
- MCR emergency habitability system (VES)
- radiation monitoring system (RMS)
- fire protection system (FPS)
- plant lighting system (ELS)

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 MCR envelope (MCRE) atmosphere are evaluated in Section 9.5.1 of this report. Manual hose stations outside the MCRE and portable fire extinguishers are provided to fight a 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. Radiation shielding corresponding to the design-basis LOCA is discussed in Section 12.3 of this report. A description of design-basis LOCA source terms and an evaluation of control room operator doses are discussed in Section 15.3 of this report. 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. Conformance with the requirements of Generic Issue B-66, “Control Room Infiltration Measurements,” and TMI Action Item III.D.3.4, “Control Room Habitability,” are discussed in Chapter 20 of this report.

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 cleanup 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 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.

The MCRE is shown in DCD Tier 2 Figures 6.4-1, 1.2-8, and 12.3-1. Areas adjacent to the MCRE are shown in DCD Tier 2 Figures 1.2-25 through 1.2-31. 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 117'-6" in the auxiliary building, on the nuclear island. As shown in 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 100'-0" is not part of the MCRE.

The VES and interfacing VBS descriptions, design parameters, instrumentation (including indications and alarms), and figures are provided in 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. Details of the radiation monitors, including testing and inspection, are provided in DCD Tier 2 Sections 7.3 and 11.5. The MCRE shielding design is evaluated in Chapter 12 of this report. The redundant, non-seismically 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 post-accident indications of VES operability are provided through the seismically qualified and non-Class 1E powered differential pressure indicators and the air flow 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, reactor cooling pump 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 function of providing passive heat sinks to limit the temperature rise in the MCR envelope, instrumentation and control rooms, and dc equipment rooms is accomplished by the VES. 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 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) due to 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 isolates 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 air flow path contains a normally open, manually operated valve to isolate and preserve the air storage tanks' contents in the event of a pressure regulating valve malfunction. The alternate air flow path contains a normally closed, manually operated valve to manually activate the alternate delivery flow path in the event the main delivery flow path is inoperable. The VES piping and penetrations for the MCR envelope 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 sub-headers 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 8895 m^3 (314,132 standard cubic feet) at a minimum pressure of 23,400 kPa (3400 psig). Each pressure regulating valve, located downstream of the common header, controls downstream pressure to approximately 790 kPa (100 psig) via 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 leak-tight 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 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 barrier design requirements and are designed to withstand specific area environment design requirements 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 eleven 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. The description of construction techniques and low leakage features to qualify the MCRE as a low-leakage boundary is provided in DCD Tier 2 Sections 3.8.4.6.1 and 6.4.2.4. Penetration sealing materials are designed to withstand at least 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 1.140-2001, 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, ASME N509 and N510 standards. RG 1.140 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 National Association standards, and duct joints are sealed with qualified non-silicone 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 off-load, SBO concurrent with a LOCA immediately following a normal refueling, and 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 is 40 °C (104 °F)) for at least 72 hours after the event and, therefore, the accessibility and equipment qualification are not challenged. DCD Tier 2 Section 6.4.2.4 states that there will be no adverse environmental effects 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. They 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 0.030 m^3/sec (65 cfm) of ventilation air to the MCRE from the air storage tanks through the main or alternate air delivery line. Westinghouse also states in the above DCD Tier 2 section that the VES flow of 0.028 m^3/sec (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, of ASHRAE Standard 62-1989, "Ventilation for Acceptable Indoor Air Quality," Appendix C. 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 in DCD Tier 2 Chapter 18.

The safety-related compressed air storage tanks are sized to provide the required air flow to the MCRE for 72 hours. During a non-radiological 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 analysis of onsite chemicals, as 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 are in accordance with RG 1.78 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 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,” to conclude 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 and (ii) for seismic Category I Class 1E toxic gas monitoring, as required and (iii) assess control room protection for toxic chemicals, and (iv) for evaluating offsite toxic releases (including the potential for toxic releases beyond 72 hours) in accordance with RG 1.78-December 2001, Revision 1, “Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release” to meet the requirements of 10 CFR Section 50.34(f)(2)(xxviii) (TMI Action Plan Item IIID.3.4) and GDC 19 ; (c) add RG 1.78-December 2001, Revision 1 to DCD Tier 2 Section 6.4.8, “References” since “Regulatory Guide 1.78-December 2001, Revision 1" replaces both “Regulatory Guide 1.78-June 1974, Revision 0" and “Regulatory Guide 1.95-January 1977, Revision 1”; (d) delete reference to “Regulatory Guide 1.95” from DCD Tier 2 Section 6.4.7; (e) revise Appendix 1A to assess the conformance with RG 1.78, Revision 1, December 2001, and revise DCD Tier 2 Tier 2 Sections 2.2, 6.4, 9.4.1, 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 TSs Bases B.3.7.6 to add a reference of 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 committed to revise DCD Tier 2 Sections 6.4.4, 6.4.7, 6.4.8, DCD Tier 2 Appendix 1A, and DCD Tier 2 Chapter 16, B3.7.6. Westinghouse incorporated these changes in Revision 4 to the DCD. However, the staff noted that the DCD still needed to be changed 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 verified that 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,” to conclude 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 is DSER confirmatory item 6.4-1.

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 1.78-2001, Revision 1.

In the text of 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 and 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 address control room protection for toxic chemicals, and evaluates 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 post-accident ESF atmospheric cleanup system and has no safety-grade source of power; therefore, it was not credited in evaluating conformance with GDC 19. Staff’s evaluation of the VBS is provided in Section 9.4 of this report.

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 153 ft-0 in, and is protected by an intake enclosure. The VBS redundant radiation monitors are located inside the MCRE. The radiation monitors and outside air isolation dampers are shown in Figure 9.4.1-1 of the DCD Tier 2. 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; and 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) 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 adequate radiation protection be provided to ensure that radiation exposures shall 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 the installation of the redundant non-safety-related supplemental air filtration units (VBS) and a safety-related emergency bottled air pressurization system (VES).

Westinghouse submitted the results of radiological consequence analyses for personnel in the MCR during a design-basis accidents in DCD Tier 2 Section 15.6.4.4. Details of the analysis assumptions for modeling the doses to MCR personnel were submitted in DCD Tier 2 Section 15.6.5.3. The staff's review of the applicant's analysis is discussed in Section 15.3 of this report.

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 AP1000 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

The staff has not completed its review of the dose to MCR personnel during the design-basis accidents at this time. The staff will complete its review once issues with the assumed aerosol removal rates in the containment, as discussed in DSER Open Item 15.3-1 have been resolved. Additionally, the staff has not completed its review of the applicant's control room atmospheric dispersion factors (see Section 2.3.4 of this report). These factors are an input to the radiological analyses. Pending resolution of the staff's concerns with the hypothetical reference control room χ/Q values, review of the control room habitability radiological consequences analyses for design basis accidents is also incomplete as discussed in DSER Open Item 15.3-2. Therefore, the resolution of issues associated with the analysis of the dose to MCR personnel during design-basis accidents is DSER Open Item 6.4-1.

The text of DCD Tier 2 Section 6.4.7 states that the 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).

The VES is tested and inspected at appropriate intervals in accordance with the surveillance and frequency requirements specified in the TS. The leak tightness 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. Air samples from the emergency air storage tanks are taken quarterly (every 92 days) and analyzed to conform with the guidelines of Table 1, and Appendix C, Table C-1 of ASHRAE Standard 62 in accordance with the TS.

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 1011 m^3 (35,700 ft^3). The maximum unfiltered air in-leakage (infiltration) into the MCRE under accident conditions is 0.00118–0.00236 m^3/sec (2.5-5.0 cfm) 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 0.0661 m^3/sec (140 cfm). The AP1000 design includes an airlock 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 once every 10 years. This testing 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 NRC staff’s RAI 410.007, the NRC staff stated that it is anticipated that the testing frequency for air in-leakage will be on the order of 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 six 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 their response to RAI 410.007 and stated that Westinghouse needs to revise their RAI 410.007 response and revise DCD Tier 2 Section 6.4.5.4 accordingly to provide an inleakage testing frequency commitment 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 their 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” and DCD Tier 2 Section 6.4.7 will be revised to add a paragraph stating that, “The Combined

License applicant will provide the testing frequency for the main control room inleakage test discussed in Section 6.4.5.4." Also DCD Tier 2 Table 1.8-2, "Summary of AP1000 Standard Plant Combined License Information Items," Item 6.4-3 is revised to refer to Section 6.4.7 for the main control room inleakage test frequency. In DCD revision 4 Westinghouse incorporated these changes. This is a COL action item and the staff finds this acceptable.

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 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 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 $^\circ$C (4.5 $^\circ$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. The staff's evaluation of the post-72 hour power supply is discussed in Section 8.3 of this report.

Preoperational testing is discussed in Chapter 14 of this report. It includes verification that a minimum VES air flow rate of 0.0307 ± 0.00236 m^3/sec (65 ± 5 scfm) will pressurize the MCRE to 3.2 mm (0.125 in) water gauge with respect to the surroundings spaces. The maximum unfiltered air in-leakage (infiltration) rate of 0.00118–0.00236 m^3/sec (2.5-5.0 cfm) during accident conditions when the VES is in operation will be verified in accordance with ASTM 741, "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 8895.2 m^3 (314,132 standard cubic feet) at a minimum pressure of 23,442 kPa (3400 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 leak tightness of the valves. Testing and inspection of the VBS safety-related radiation monitors are discussed in Section 11.5 of this report. The air quality within the MCR/TSC environment is confirmed to be within the guidelines of Table C-1 of ASHRAE Standard 62 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 are listed in DCD Tier 2 Table 6.4-2 and are located in the MCR. Actuation and radiation monitoring instrumentation for the VBS and VES are discussed in Sections 7.3 and 11.5 of this report.

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 Sections 3.3, 3.6, and 3.8 of the DCD Tier 2. Westinghouse also stated that the flood protection measures for seismic Category 1 SSCs are designed in accordance with RG 1.102 and RG 1.59. 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 either are 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, "Environmental and Dynamic Effects Design Bases," 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 GDCs 4, 5, and 19, as referenced in Section 6.4 of the SRP and consequently with the subject SRP acceptance criteria. The staff is unable to reach a conclusion because insufficient information was available at the time of this review. Westinghouse needs to resolve Confirmatory Item 6.4-1 concerning the toxic assessment for the control room habitability; and Open Item 6.4-1 concerning independent analysis of the dose to the MCR personnel during a design-basis LOCA.

<u>Control Room Habitability and Toxic Chemicals</u>

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 6.4-1.

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 non-gaseous 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 passive core cooling system discussed in DCD Tier 2 Section 6.3, "Passive Core Cooling System." The fire protection system provides a non-safety-related containment spray function for accident management following a severe accident. This design is not credited in any analysis. Natural mechanisms for removal of airborne activity are discussed further in 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."

The AP1000 design does include a non-safety-related containment spray which is part of the fire protection system 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. Evaluation of the non-safety-related containment spray system is in Section 19.2.3.3.9 of this report.

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. 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 is discussed in DCD Tier 2 Appendix 15B. No active fission product control systems are required in the AP1000 design to meet the regulatory requirements. These natural fission product control mechanisms and the limited containment leakage result in offsite doses that are less than those specified in 10 CFR Section 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, GDC 39, "Inspection of Containment Heat Removal," GDC 40, "Testing of Containment Heat Removal," GDC 42, "Inspection of Containment Atmosphere Cleanup Systems," 43, "Testing of Containment Atmosphere Cleanup Systems," 45, "Inspection of Cooling Water System," and GDC 46, "Testing of Cooling Water System," are specified, in part, in 10 CFR Part 50, Section 50.55a, "Codes and Standards," and detailed in Section XI of the ASME Code.

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

1. Components subject to inspection

The applicant's definition of 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 the ASME Code, Section III, Article NCA-2000.

2. Accessibility

10 CFR 50.55a(g)(3)(ii) requires that ASME Code Class 2 and Class 3 components and supports be designed and provided with access to enable the performance of inservice examination of such components and 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 the following considerations: (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; (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 in agreement with the requirements of IWA-2000, IWC-2000, and IWD-2000 of Section XI of the ASME Code.

4. Evaluation of Examination Results

The methods for evaluation of the results are acceptable if in agreement with the requirements of IWC-3000 and IWD-3000 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 of Section XI of the Code.

6. Augmented ISI to Protect Against Postulated Piping Failure

High-energy fluid piping between containment isolation valves receive an augmented 100 percent volumetric examination of circumferential and longitudinal pipe welds.

7. GDC

Compliance with the preservice and inservice examination requirements of 10 CFR Part 50, Section 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 ISI program for 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, in part, 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, in part, 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.

GDC 43 requires, in part, that the containment atmosphere cleanup systems be designed to permit periodic pressure testing to assure the structural and leaktight integrity of their components.

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.

GDC 46 requires, in part, that the cooling system be designed to permit periodic pressure testing to assure the structural and leaktight integrity of its components.

Summary Technical Information

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

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 is to be delineated in the inspection program. 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. The requirements for system pressure tests and visual examinations are defined in Section XI, IWA-5000. 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 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 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 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. 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 post design certification activities. The preservice/ISI programs will comply with applicable provisions of 10 CFR 50.55a(b)(2).

<u>Staff Evaluation</u>

The staff evaluation of ISI of 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 the ASME Code, Section III, as required by 10 CFR 50.55a and thus Class 2 and 3 components subject to inspection are in agreement with definitions acceptable to the staff in Code Section III, Article NCA-2000.

2. Accessibility

The AP1000 design follows the ASME Code provisions for accessibility which include the following considerations: (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 the ASME Code, Section XI, Section IWA-1500, Accessibility, and thus meet requirements acceptable to the staff with respect to accessibility.

The staff reviewed the DCD Tier 2 Section 6.6 to assure that compliance with the regulations for the design of the 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 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 so that the ASME Code Class 2 and 3 components are designed to allow for the application of the required ISI methods, that is, 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. The staff experience is that 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, lighting, installation of 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 unanticipated 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 the ASME Code, Section XI, as required by 10 CFR 50.55a and 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.

4. Evaluation of Examination Results

The ISI program will follow the ASME Code, Section XI as required by 10 CFR 50.55a and 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.

5. System Pressure Tests

The ISI program will follow the ASME Code, Section XI as required by 10 CFR 50.55a and thus the system pressure testing will meet requirements acceptable to the staff in IWA-5000 of Section XI of the Code.

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 where 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. Since the proposed program satisfies the criteria of SRP 6.6, the staff finds this augmented ISI program acceptable.

7. GDC

For the AP1000 Design, the applicability of the GDCs was reviewed. 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 Class 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 GDCs 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 the DCD Tier 2 Section 6.3. The staff's evaluation of the use of the PXS in lieu of an ECCS is in Section 6.3 of this report. This system is safety-related and contains ASME Code Class 1, 2, and 3 components. As such, this system is subject to 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 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. The staff's evaluation of the PCS is discussed in Section 6.2.2 of this report. This system is safety-related and contains ASME Code Class 3 components. As such, this system is subject to 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 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 post-accident 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 fire protection system that provides for non-safety-related containment spray function for severe accident management includes equipment and valves such as the fire pumps and fire main header. The staff's evaluation of the containment spray system is provided in Section 6.5.2 and Chapter 19 of this report. 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 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 passive core cooling system and the passive containment cooling system. The staff's evaluation of the component cooling water system is discussed in Section 9.2.2 of this report. 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. Those portions 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 passive systems discussed above. Therefore, GDC 45 and 46 are not applicable to the AP1000 design.

Conclusion

The staff concludes that the AP1000 ISI program for 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.